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TOKYO, JAPAN

August 22, 2008

Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Attention: Mr. Jeffrey A. Ciocco

Docket No. 52-021
MHI Ref: UAP-HF-08141

Subject: MHI's Response to NRC's Request for Additional Information on US-APWR Topical Report MUAP-07010-P, Non-LOCA Methodology

Mitsubishi Heavy Industries, Ltd. ("MHI") transmits to the U.S. Nuclear Regulatory Commission ("NRC") the documents entitled "MHI's Response to NRC's Request for Additional Information on US-APWR Topical Report MUAP-07010-P, Non-LOCA Methodology" and "Supplemental Material Referenced in the Response to the Request for Additional Information." In the enclosed materials MHI provides responses to the NRC's "Request for Additional Information ("RAI") on US-APWR Topical Report MUAP-07010-P, Non-LOCA Methodology," dated July 16, 2008. As agreed upon during the June 26, 2008 conference call, the responses to questions 2.1-16, 3.2-2, 3.2-7, 3.2-10, 5.5-2 will be provided within 60 days from the date of the RAIs; all other responses are included in the enclosed document.

As indicated in the enclosed materials, this document contains information that MHI considers proprietary or has been provided to MHI under the obligation to maintain its confidentiality, and therefore should be withheld from disclosure pursuant to 10 C.F.R. § 2.390 (a)(4) as trade secrets and commercial or financial information which is privileged or confidential. A non-proprietary version of one of the documents is also being submitted in this package (Enclosure 3). In the non-proprietary version, the proprietary information, bracketed in the proprietary version, is replaced by the designation "[ ]".

This letter includes a copy of the proprietary version of the RAI response (Enclosure 2), a copy of the non-proprietary version of the RAI response (Enclosure 3), proprietary supporting material referenced in the RAI response (Enclosure 4), and the Affidavit of Yoshiki Ogata (Enclosure 1) which identifies the reasons MHI respectfully requests that all materials designed as "Proprietary" in Enclosures 2 and 4 be withheld from disclosure pursuant to 10 C.F.R. § 2.390 (a)(4).

Please contact Dr. C. Keith Paulson, Senior Technical Manager, Mitsubishi Nuclear Energy Systems, Inc., if the NRC has questions concerning any aspect of this submittal. His contact information is provided below.

Sincerely,

Y. Ogata

Yoshiki Ogata
General Manager- APWR Promoting Department
Mitsubishi Heavy Industries, Ltd.

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Enclosures:

1. Affidavit of Yoshiki Ogata
2. MHI's Responses to NRC's Request for Additional Information on US-APWR Topical Report MUAP-07010-P, Non-LOCA Methodology (proprietary)
3. MHI's Responses to NRC's Request for Additional Information on US-APWR Topical Report MUAP-07010-P, Non-LOCA Methodology (non-proprietary)
4. Supplemental Material Referenced in the Response to the Request for Additional Information (proprietary)

CC: J. A. Ciocco
C. K. Paulson

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ENCLOSURE 1

Docket No.52-021

MHI Ref: UAP-HF-08141

MITSUBISHI HEAVY INDUSTRIES, LTD.

AFFIDAVIT

I, Yoshiki Ogata, being duly sworn according to law, depose and state as follows:

1. I am General Manager, APWR Promoting Department, of Mitsubishi Heavy Industries, Ltd. ("MHI"), and have been delegated the function of reviewing MHI's US-APWR documentation to determine whether it contains information that should be withheld from disclosure pursuant to 10 C.F.R. § 2.390 (a)(4) as trade secrets and commercial or financial information which is privileged or confidential.

2. In accordance with my responsibilities, I have reviewed the enclosed documents entitled "MHI's Response to NRC's Request for Additional Information on US-APWR Topical Report MUAP-07010-P, Non-LOCA Methodology", dated August 22, 2008 and "Supplemental Material Referenced in the Response to the Request for Additional Information," dated August 22, 2008, and have determined that the documents contain proprietary information that should be withheld from public disclosure. Those pages containing proprietary information are identified with the label "Proprietary" on the top of the page and the proprietary information has been bracketed with an open and closed bracket as shown here "[ ]". The first page of each of the documents indicates that all information identified as "Proprietary" should be withheld from public disclosure pursuant to 10 C.F.R. § 2.390 (a)(4).

3. The basis for holding the referenced information confidential is that it either describes the unique design of the safety analysis, developed by MHI (the "MHI Information"), or includes information that was provided to MHI pursuant to licensing agreements with a third party (the "Licensor") for MHI's use and under the obligation to maintain its confidentiality (the "Licensed Information").

4. The MHI Information is not used in the exact form by any of MHI's competitors. This information was developed at significant cost to MHI, since it required the performance of research and development and detailed design for its software and hardware extending over several years. Furthermore, MHI has an ownership interest in the Licensed Information by having paid significant sums of money to the Licensor for the rights to the intellectual property therein such that public disclosure of the materials would adversely affect MHI's competitive position.

5. The referenced information has in the past been, and will continue to be, held in confidence by MHI and and is always subject to suitable measures to protect it from unauthorized use or disclosure.

6. The referenced information is not available in public sources and could not be gathered readily from other publicly available information.

7. The referenced information is being furnished to the Nuclear Regulatory Commission

("NRC") in confidence and solely for the purpose of supporting the NRC staff's review of MHI's Application for certification of its US-APWR Standard Plant Design.

8. Public disclosure of the referenced information would assist competitors of MHI and the Licensor in their design of new nuclear power plants without the costs or risks associated with the design and testing of new systems and components. Disclosure of the information identified as proprietary would therefore have negative impacts on the competitive position of MHI in the U.S. nuclear plant market.

I declare under penalty of perjury that the foregoing affidavit and the matters stated therein are true and correct to the best of my knowledge, information, and belief.

Executed on this 22th day of August, 2008.

Y. Ogata

Yoshiki Ogata

Enclosure 3

UAP-HF-08141
Docket No. 52-021

August 22, 2008

MHI's Response to NRC's Request for Additional Information on US-APWR Topical Report MUAP-07010-P, Non-LOCA Methodology

(Non-Proprietary)

RAI 2.1-1

The evolution of MARVEL-M 4-loop code from the MARVEL 2-loop code was discussed. Please provide details of the significant modeling changes between the two codes.

Response

Section 2.1.3 of MUAP-07010 describes the main differences between the 2-loop MARVEL and 4-loop MARVEL-M codes. The code improvements incorporated into MARVEL-M with the greatest potential impact on the safety analysis methodology are the expansion from 2-loop to 4-loop simulation and the addition of a built-in reactor coolant pump (RCP) model. MUAP-07010 Section 2.1.3.1 provides a detailed discussion of the 4-loop MARVEL-M reactor coolant system model, including schematics of the overall plant and reactor vessel flow models. Section 2.1.3.2 provides a detailed discussion of the mixing equations utilized in MARVEL-M. Section 2.1.3.3 describes the new reactor coolant pump model. The loop noding, transport, and mixing models, as well as the steam generator and core heat transfer models remain essentially the same as in the original MARVEL code.

The following changes from the original version of MARVEL are briefly described in Section 2.1.3.5.

- Pressurizer surge line model
- Hot-spot fuel thermal kinetics model
- Core void simulation
- Feedline break blowdown simulation
- Conversion of RCS volume balance by pressure search

These last five changes are characterized as model refinements in MUAP-07010 rather than significant code changes and have a minimal impact on the safety analysis methodology.

RAI 2.1-2

Please provide a description of the simplified DNBR model in MARVEL-M.

Response

MARVEL-M calculates the value of DNBR during a transient using a simplified DNBR lookup table methodology.

The MARVEL-M lookup table is an array or database of DNBR, core inlet temperature, pressure, and core heat flux data that is created by VIPRE-01M steady state calculations. VIPRE-01M calculates the core inlet temperature corresponding to a given set of parameters such as pressure, core heat flux and DNBR under the steady-state assumptions of constant core flow rate and constant core power distribution.

This simplified DNBR model is used for the evaluation of events which have constant core flow rate and are bounded by the applicable power distribution. For other events whose parameters exceed the limitations of the lookup table, VIPRE-01M is used to directly calculate DNBR based on a DNB correlation.

Figure 2.1-2.1 Schematic Description of the Simplified DNBR Model in MARVEL-M

RAI 2.1-3

Have any MARVEL/MARVEL-M code comparisons been performed? Please provide a MARVEL/MARVEL-M code comparison for a typical 2-loop calculation.

Response

Comparisons between the MARVEL and MARVEL-M codes have been performed for non-LOCA events as a verification of the extension from a 2-loop simulation to a 4-loop simulation. The results of these comparisons are introduced in the following figures. Figures 2.1-3.1 through 2.1-3.6 show the comparison of "Loss of Load" as a uniform event and Figures 2.1-3.7 through 2.1-3.17 show the comparison of "Feedwater System Pipe Break" as a non-uniform event. The 4-loop MARVEL-M case is set up to assume the break in a single loop with all of the remaining loops receiving the same emergency feedwater flow. This allows a direct comparison with 2-loop MARVEL, which models these loops as a single loop. Both the 2-loop MARVEL and 4-loop MARVEL-M runs utilize the hyperbolic coast down reactor coolant pump model, again allowing for direct comparison of the code output.

The results demonstrate nearly exact agreement between the 2-loop and 4-loop codes for both primary and secondary parameters of interest.

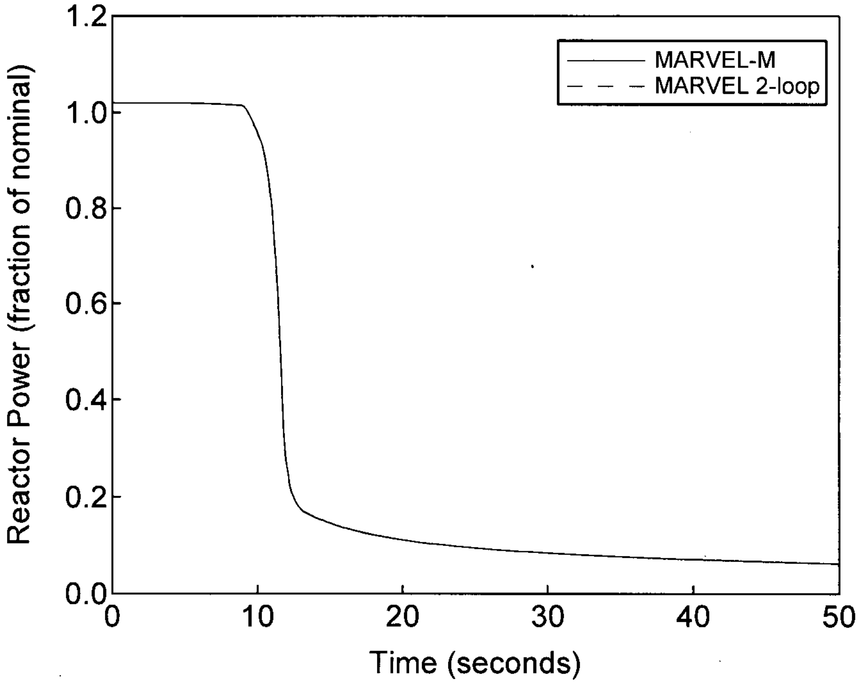


Figure 2.1-3.1 **Reactor Power versus Time**
Loss of Load

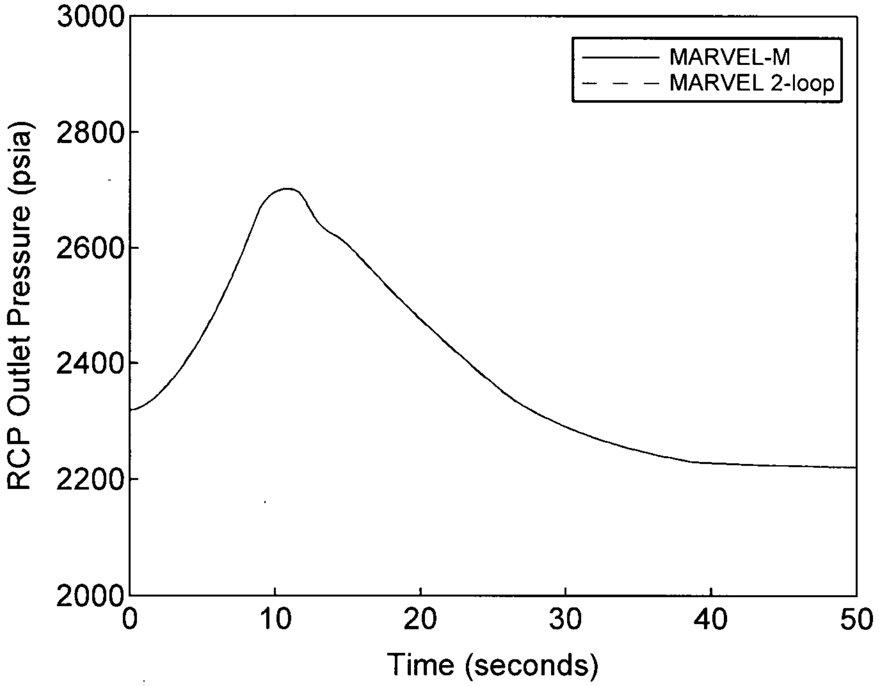


Figure 2.1-3.2 **RCP Outlet Pressure versus Time**
Loss of Load

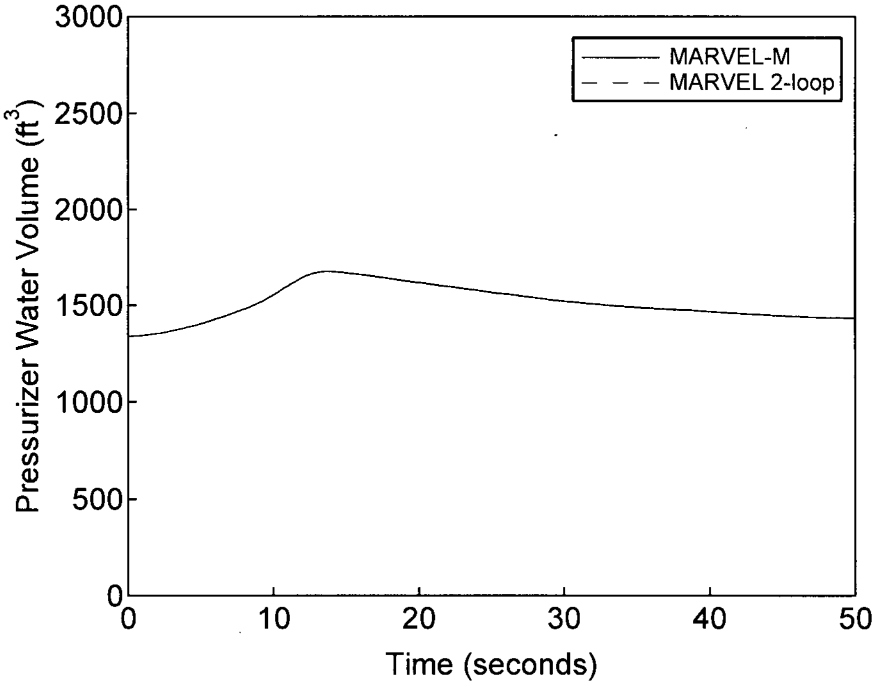


Figure 2.1-3.3 Pressurizer Water Volume versus Time Loss of Load

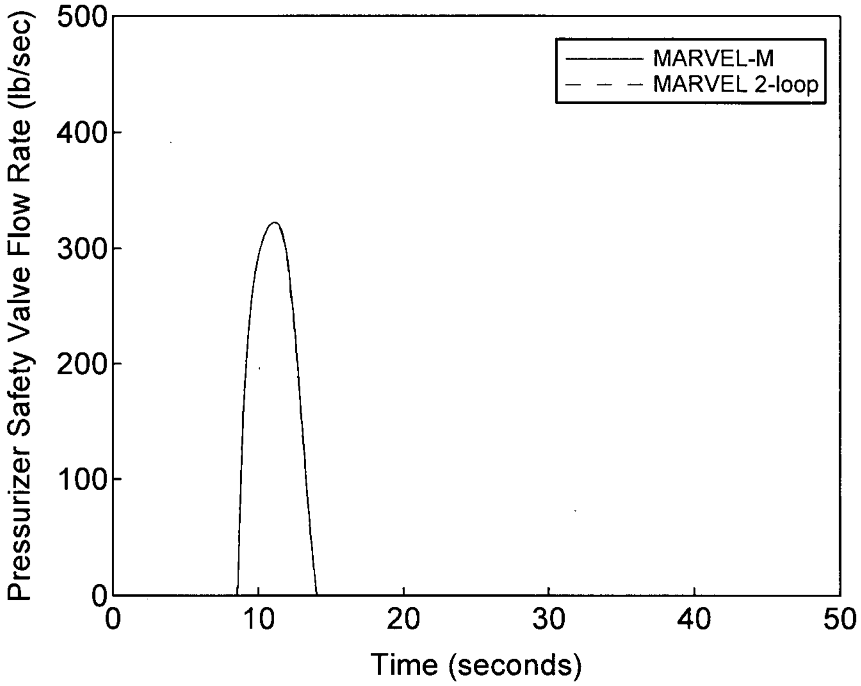


Figure 2.1-3.4 Pressurizer Safety Valve Flow Rate versus Time Loss of Load

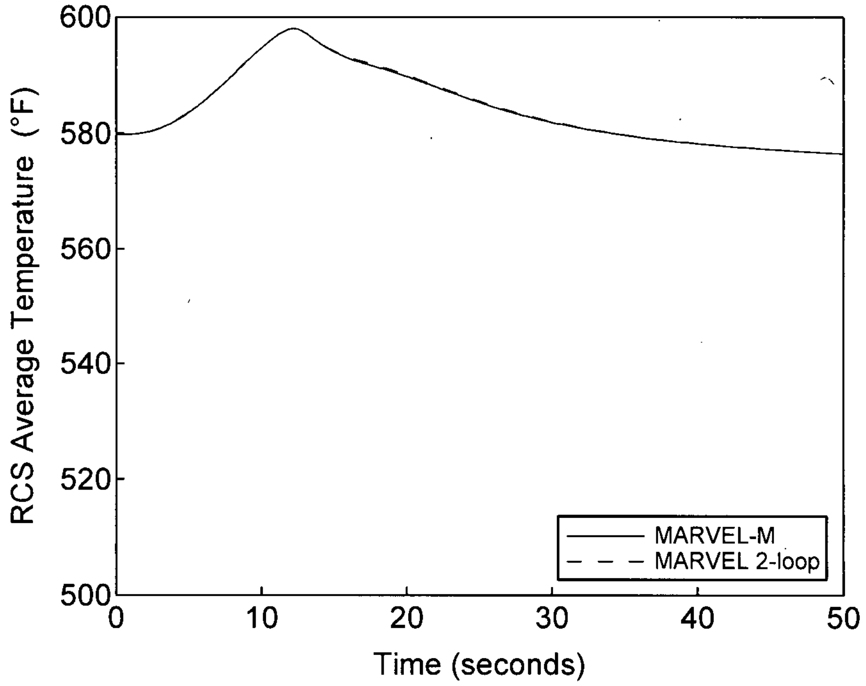


Figure 2.1-3.5 RCS Average Temperature versus Time
Loss of Load

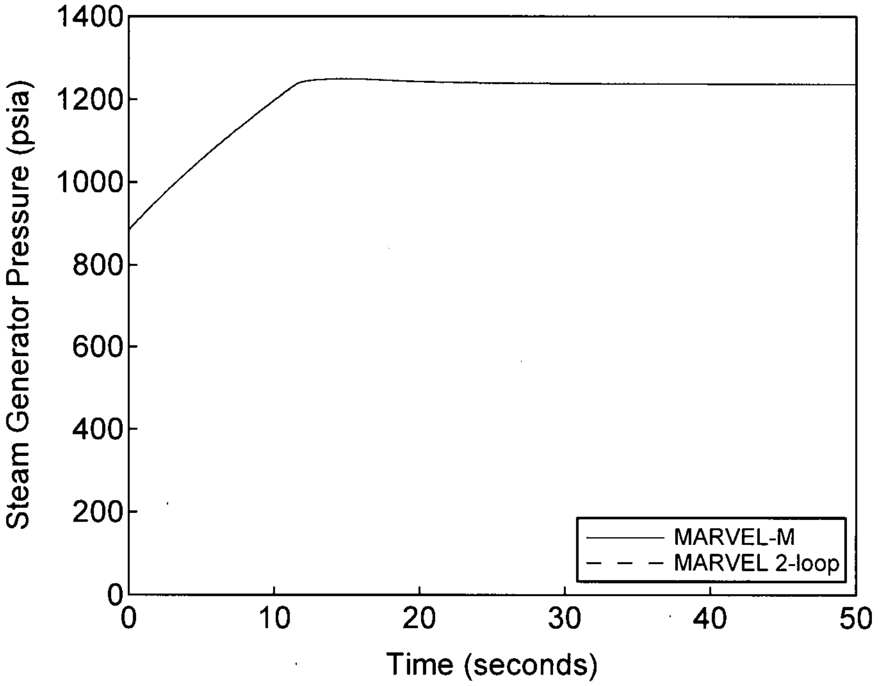


Figure 2.1-3.6 Steam Generator Pressure versus Time
Loss of Load

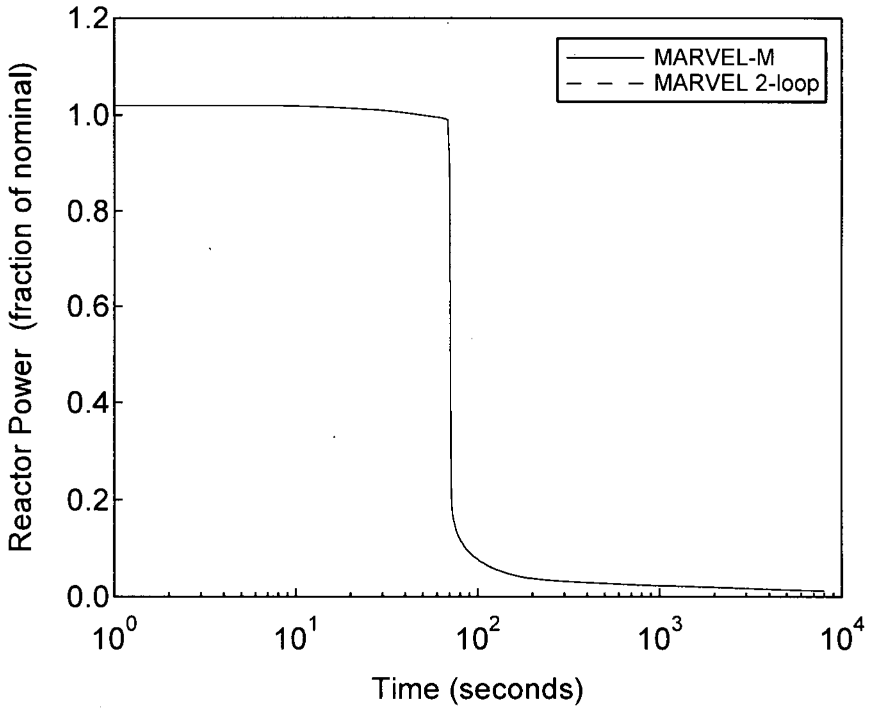


Figure 2.1-3.7 Reactor Power versus Time
Feedwater System Pipe Break

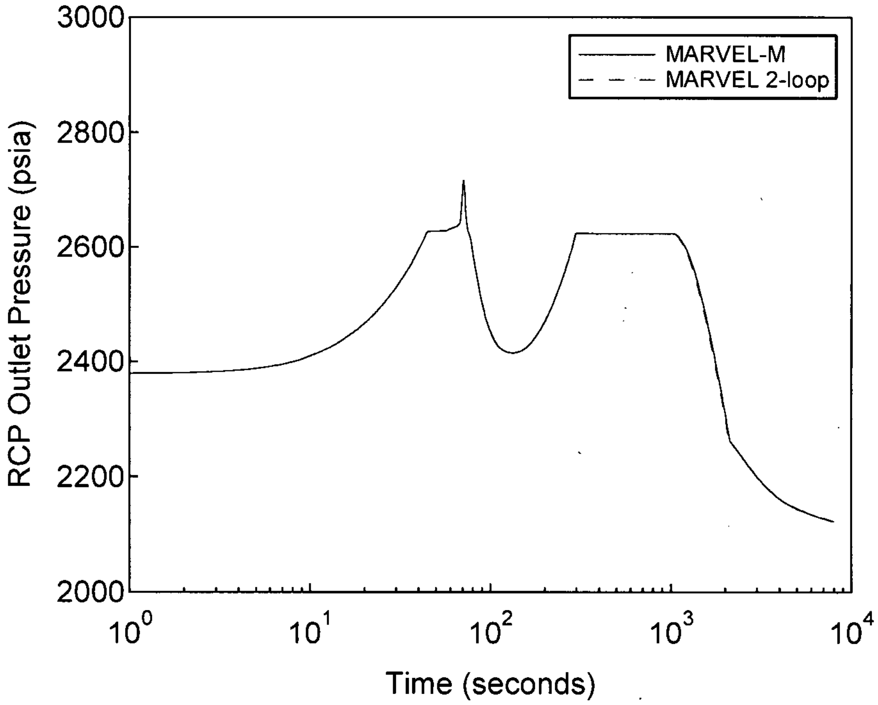


Figure 2.1-3.8 RCP Outlet Pressure versus Time
Feedwater System Pipe Break

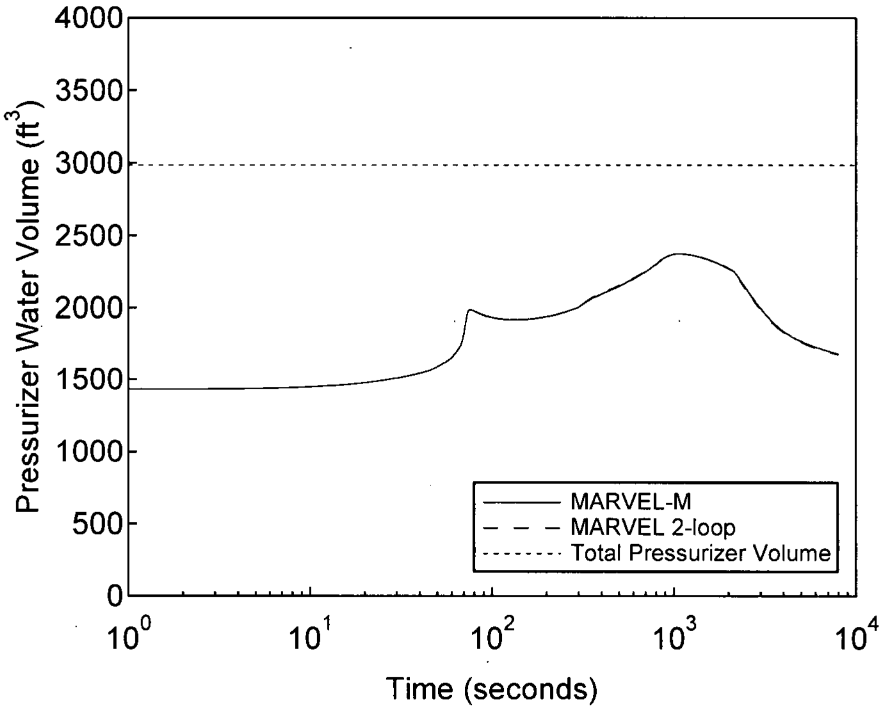


Figure 2.1-3.9 **Pressurizer Water Volume versus Time Feedwater System Pipe Break**

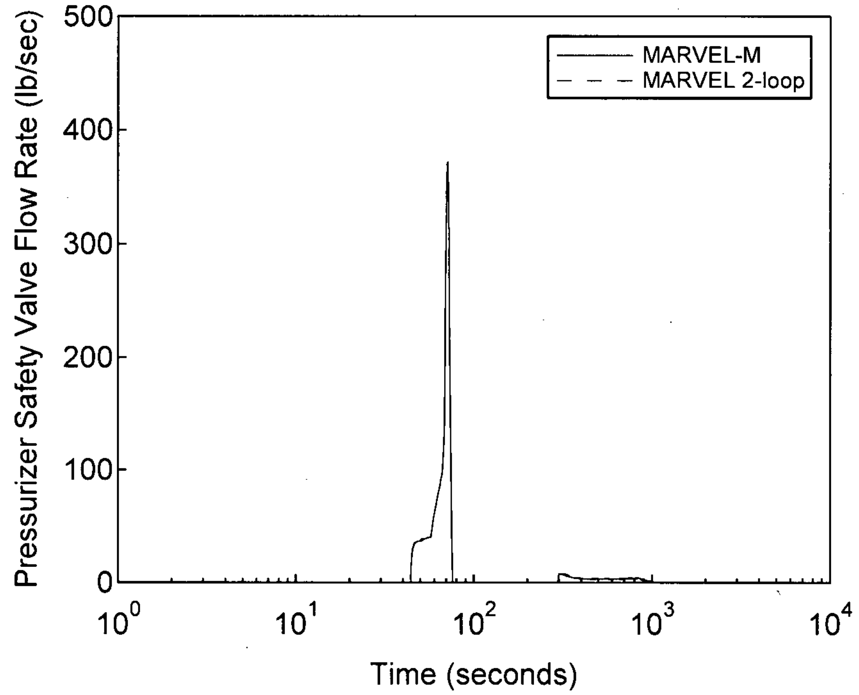


Figure 2.1-3.10 **Pressurizer Safety Valve Flow Rate versus Time Feedwater System Pipe Break**

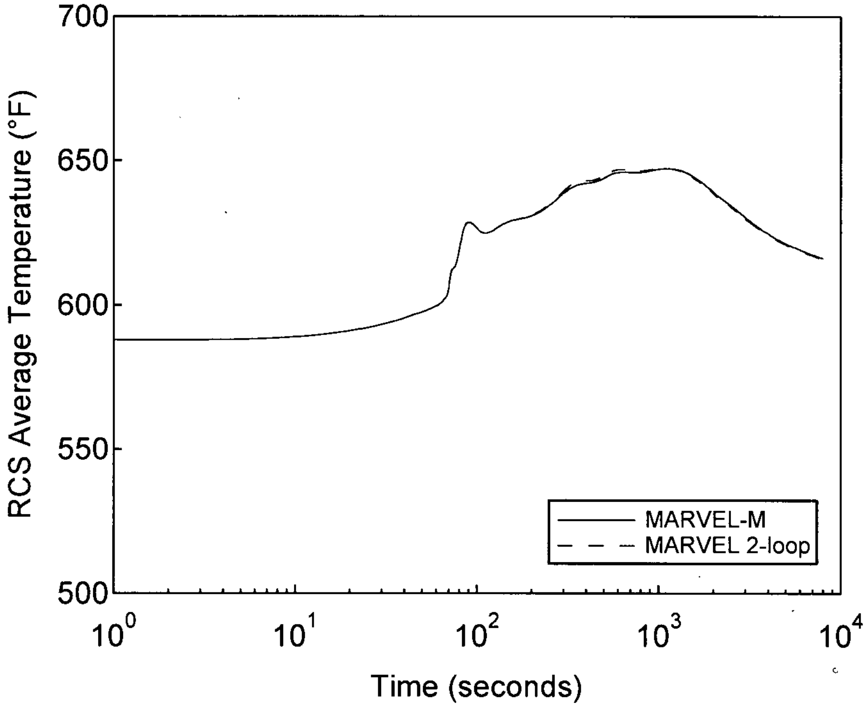


Figure 2.1-3.11 RCS Average Temperature versus Time
Feedwater System Pipe Break

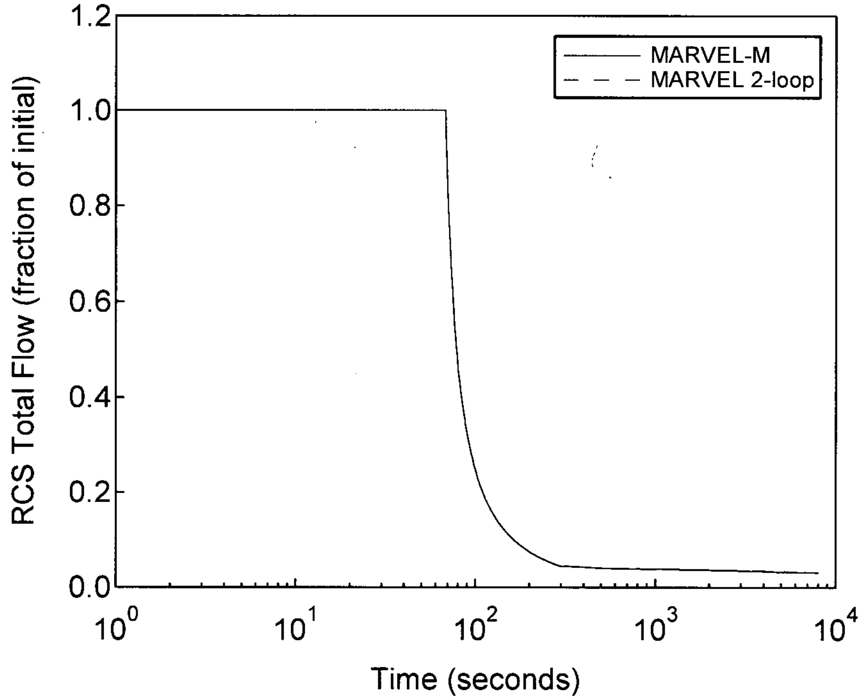


Figure 2.1-3.12 RCS Total Flow versus Time
Feedwater System Pipe Break

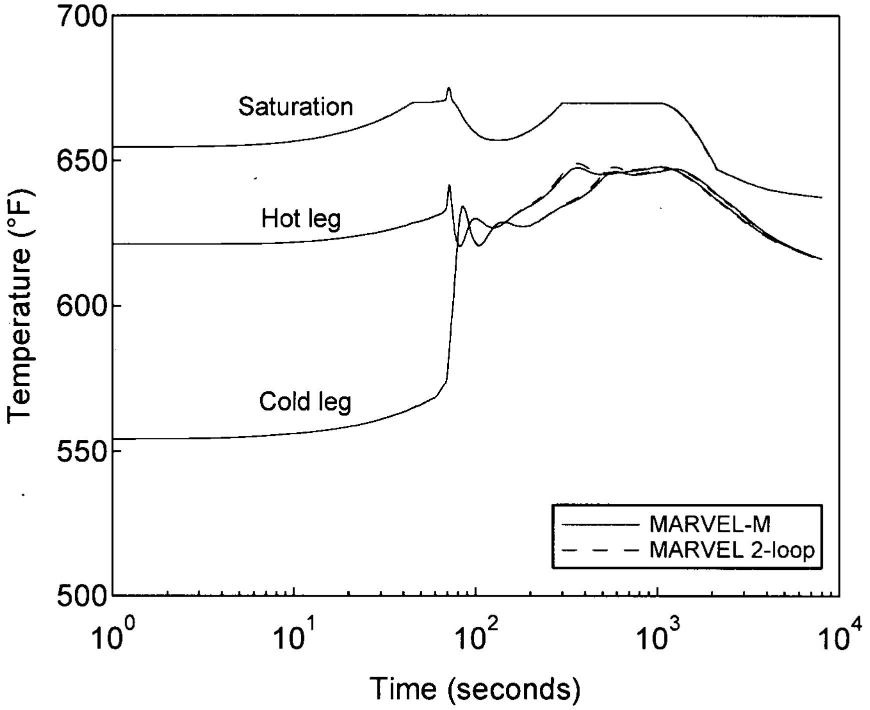


Figure 2.1-3.13 **Temperature of Faulted Loop versus Time Feedwater System Pipe Break**

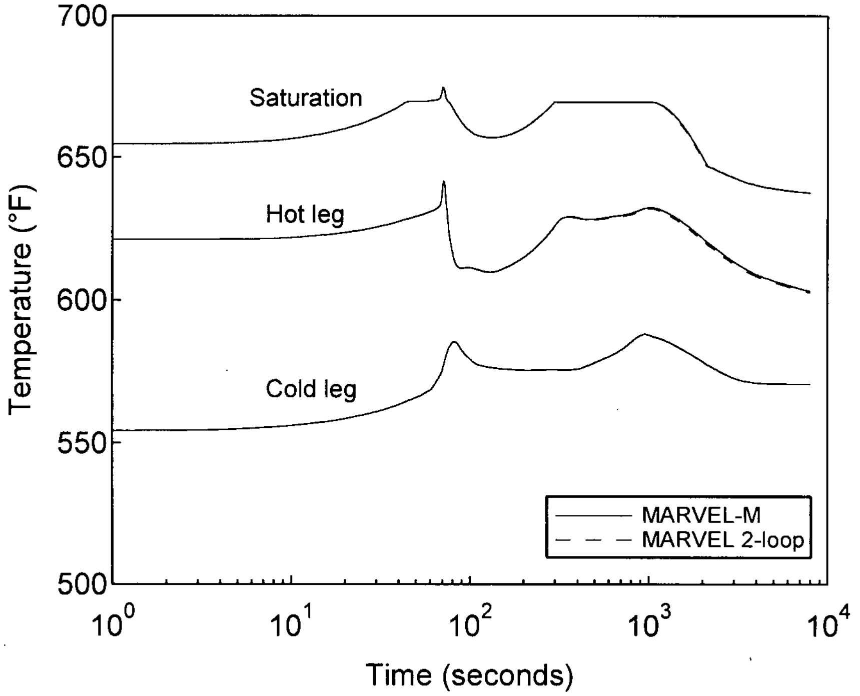


Figure 2.1-3.14 **Temperature of Intact Loop versus Time Feedwater System Pipe Break**

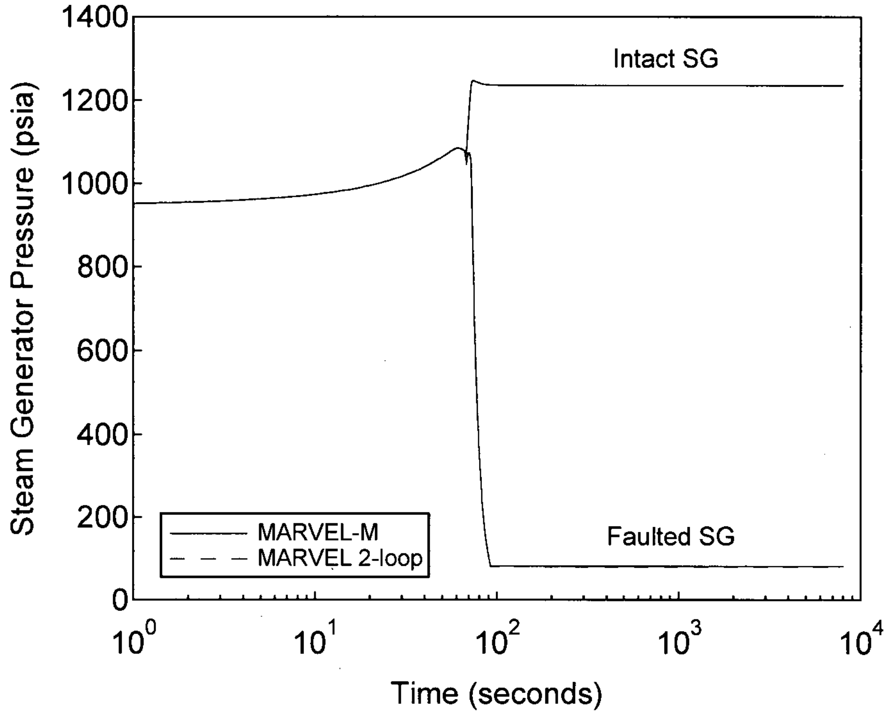


Figure 2.1-3.15 Steam Generator Pressure versus Time
Feedwater System Pipe Break

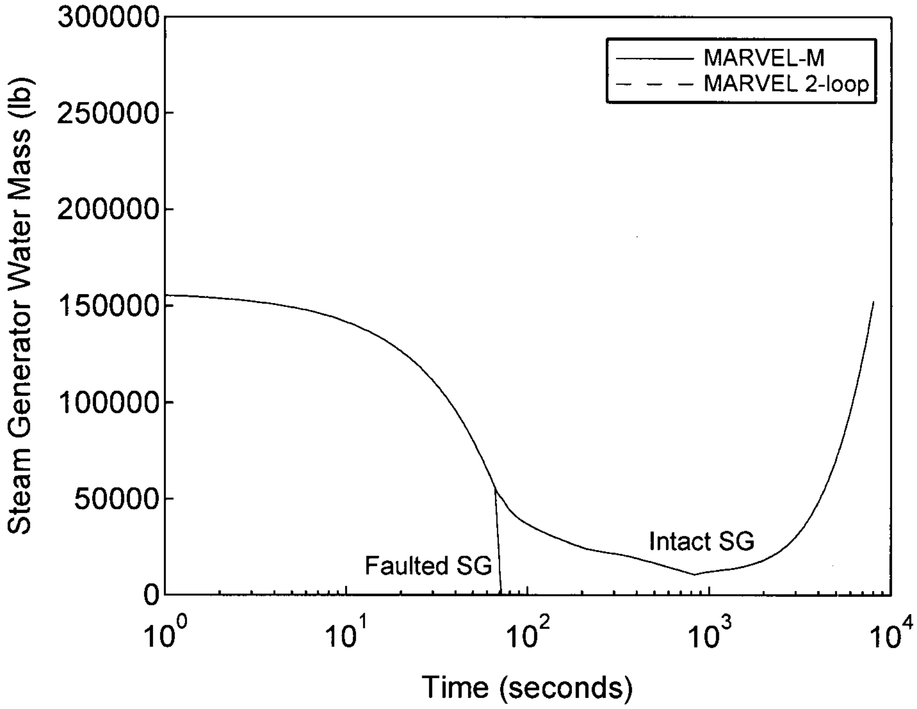


Figure 2.1-3.16 Steam Generator Water Mass versus Time
Feedwater System Pipe Break

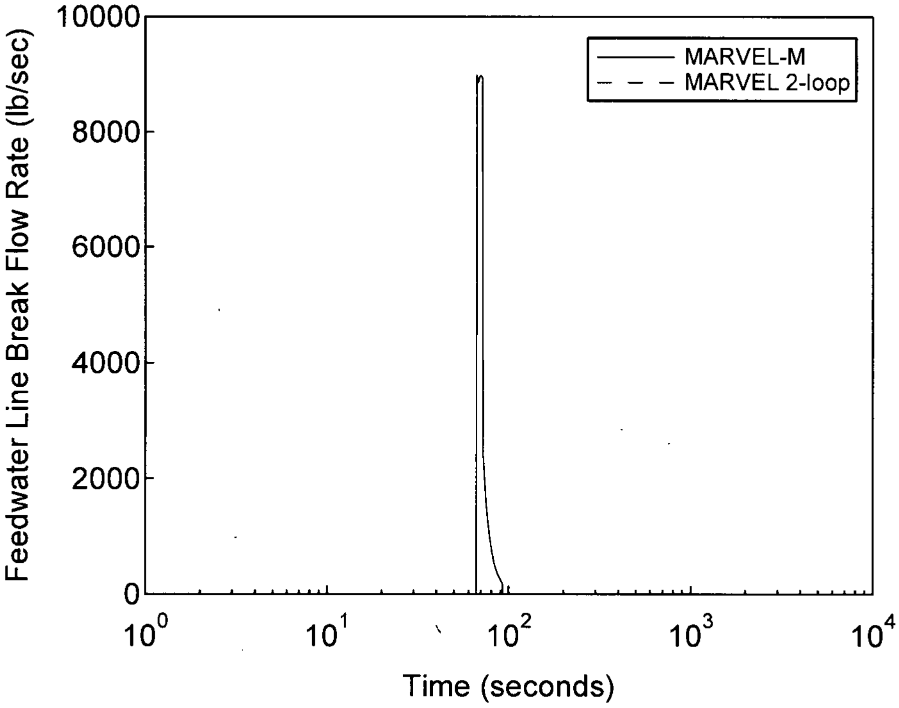


Figure 2.1-3.17 **Feedwater Line Break Flow Rate versus Time Feedwater System Pipe Break**

RAI 2.1-4

If two-phase homogeneous flow is not applicable, what model does the code use for two-phase flow for $\alpha > \alpha_{homogeneous}$? How does the user deal with conditions in which homogeneous two-phase flow is not applicable?

Response

For the majority of non-LOCA events analyzed in the US-APWR Chapter 15 analysis, no voids exist in the RCS except in the pressurizer and reactor vessel head. Thus, boiling beyond homogeneous two-phase flow does not occur for non-LOCA analyses which use the MARVEL-M code. MARVEL-M prints a message to inform the user of the occurrence of boiling. The user must determine if the predicted volumetric void fraction has a significant impact on the validity of the computed results.

Localized voids exist for the radiological analysis case of the steam generator tube rupture (SGTR) event. If the pressurizer becomes empty, short-term boiling (~20 seconds) occurs in the hot leg volume where the pressurizer surge line is connected. This phenomenon does not occur in the loops that are not connected to the pressurizer and is not a direct effect of changes in temperature or pressure associated with the reactor coolant system. Since the subcooling in the hot leg is not enough to condense all of the voids from the pressurizer, small portions of the voids remain in the hot leg. The quality at the node connecting the surge line to the hot leg is approximately 0.5% with a corresponding void fraction of about 10%. The voids flow from the hot leg to the steam generator where they are condensed to liquid. A void fraction of 10% corresponds to a dispersed bubbly flow regime, which can be treated as homogeneous two-phase flow for the short-duration of its occurrence during this event. Therefore, it is acceptable to apply MARVEL-M to the SGTR analysis.

In summary, a message in the MARVEL-M output alerts the user to the occurrence of RCS boiling, and a case-by-case review is performed to verify whether the existing result is valid or if an alternative course of action is needed.

RAI 2.1-5

Have any of the mixing models been compared to scaled tests? If so, please provide the comparisons.

Response

The MARVEL-M code inputs derived from the scaled test results are described in the response to RAI 2.1-13.

RAI 2.1-6

Please provide the methodology for the natural circulation flow modeling.

Response

Section 2.1.3.3 of MUAP-07010 describes the MARVEL-M reactor coolant pump (RCP) and flow transient models. MARVEL-M includes an RCP model so that flow transients can be explicitly computed by the pump model in conjunction with the existing reactor coolant system hydraulic models. See the response to RAI 2.1-18 for additional details regarding the use of MUAP-07010 Equation (36) to determine the loop flow for both pump coastdown and natural circulation conditions.

RAI 2.1-7

Please provide the details of the reactor coolant pump model.

Response

MUAP-07010 Section 2.1.3.3 and the response to RAI 2.1-18 provide details regarding the reactor coolant pump and flow transient models utilized in MARVEL-M. Additional details regarding these models are also available in GEN0-LP-480 (the MARVEL-M code manual) Section 1.7. Revision 3 of GEN0-LP-480 was submitted to the NRC by MHI letter UAP-HF-08138 on August 1, 2008.

The original MARVEL code was designed to accept input loop flow tables as a function of time prepared by off-line computation. MARVEL-M now includes a reactor coolant pump (RCP) model so that flow transients can be internally calculated by the pump model in conjunction with the existing reactor coolant system (RCS) hydraulic models. The fundamental flow transient equations in MARVEL-M are based on a momentum balance around each reactor coolant loop and across the reactor vessel, combined with flow continuity and the RCP momentum balance and operational characteristics. Transient operating conditions for the RCP model are determined from homologous head and hydraulic torque curves and pump motor speed-torque characteristics input to MARVEL-M. The overall RCP model methodology described below is similar to the models employed in existing NRC approved codes such as LOFTRAN and PHOENIX.

MUAP-07010 Equation (36) provides the set of simultaneous equations used to describe the overall RCS hydraulic model utilized in MARVEL-M to calculate changes in loop flow. The term H_{pump} in this equation represents the RCP head. Thus, the method of calculating a value of H_{pump} for use in Equation (36) constitutes the RCP model utilized in MARVEL-M.

The user can provide input values for variables defining pump characteristics, homologous head-speed-flow curve and torque-speed-flow curve input, reference pump parameters, and options for pump control. (See GEN0-LP-480 Part II Section 2.5 for a detailed listing of specific MARVEL-M input parameters.)

The pump speed is determined from a torque balance on each RCP. The pump motor torque must equal the sum of the various torques developed in the RCP, as mathematically described below (MUAP-07010 Equation 28).

$$T_M = T_{KE} + T_H + T_W + T_R$$

where

T_M = pump motor torque calculated from the speed-torque curve input to MARVEL-M

T_{KE} = pump kinetic torque = $\frac{1}{g} I_p \cdot \frac{d\omega}{dt}$, where g is the gravitational constant, I_P is the moment of inertia, and ω is the angular speed of rotation

T_H = pump hydraulic torque calculated from the torque-speed-flow homologous curve input to MARVEL-M

T_W = pump windage and friction torque = $K_W \cdot \omega^n$, where K_W and n are constants input to MARVEL-M

T_R = retardation torque due to eddy current in the RCP motor stator

This equation can be rearranged to provide the following equation of motion that is solved to give the transient pump speed:

$$\frac{d\omega}{dt} = \left(T_M - T_H - K_W \cdot \omega^n - T_R\right) \cdot \frac{g}{I_p}$$

Once the pump speed has been determined, the pump head is interpolated from the homologous head-speed-flow curves and de-normalized based on reference parameters that are separately input to MARVEL-M. Then the pump head is used to re-calculate the loop flow, which ultimately feeds back into the calculation of a new pump speed at the next time step. Note that T_R is actually treated as part of T_M.

RAI 2.1-8

Describe the modeling of the four major thermal resistances in the calculation of the overall heat transfer coefficient for the steam generators.

Response

The steam generator heat transfer coefficient model is described in detail in Section 1.5.1 of GEN0-LP-480 (the MARVEL-M code manual). Revision 3 of this document was submitted to the NRC by MHI letter UAP-HF-08138 on August 1, 2008.

The overall heat transfer coefficient, U, in the steam generators consists of four major thermal resistance components: the primary convection film resistance, the tube metal resistance, the fouling resistance, and the secondary side boiling heat transfer resistance.

$$U = \frac{1}{R_{tot}} = \frac{1}{\left(R_{pf} + R_{tube} + R_{foul} + R_{bo}\right)}$$

where

$$R_{pf} = R_{pf}{}^{o} \frac{(1+10^{-2}T_0 - 10^{-5}T_0{}^2)}{(1+10^{-2}T - 10^{-5}T^2)} \left(\frac{Q}{Q^0}\right)^{-0.8},$$

$$R_{tube} = R_{tube}{}^{0} \frac{8.0 + 0.0051 \cdot T_{tube}{}^{0}}{8.0 + 0.0051 \cdot T_{tube}},$$

$$R_{bo} = R_{bo}{}^{0} \left(\frac{\Delta T}{q''}\right) / \left(\frac{\Delta T}{q''}\right)^{0}, \text{ and}$$

R_{foul} = constant during a transient.

From these equations, it is observed that the absolute value of the overall steam generator heat transfer coefficient at nominal conditions is determined within the MARVEL-M code using the nominal full thermal power, nominal average coolant temperature, nominal steam temperature, and nominal ΔT.

Thus, only the ratios of the major thermal resistances as a fraction of the overall primary-to-secondary heat transfer resistance at nominal full power conditions are directly input to MARVEL-M. The MARVEL-M input parameter variables are provided below with the equations that mathematically describes the ratios in terms of the power, coolant temperature, etc.

$$\text{RRPF} = \frac{R_{pf}}{R_{tot}{}^{0}} = \left(\frac{R_{pf}{}^{0}}{R_{totR}{}^{0}}\right) \cdot \left(\frac{1+10^{-2}T_0 - 10^{-5}T_0^2}{1+10^{-2}T - 10^{-5}T^2}\right) \left(\frac{Q}{Q_0}\right)^{-0.8}$$

$$\text{RRTM} = \frac{R_{tube}}{R_{tot}{}^{0}} = \left(\frac{R_{tube}{}^{0}}{R_{tot}{}^{0}}\right) \frac{8.0 + 0.0051 \cdot T_t^0}{8.0 + 0.0051 \cdot T_t}$$

$$\text{RRBO} = \frac{R_{bo}}{R_{tot}{}^{0}} = \left(\frac{R_{bo}{}^{0}}{R_{tot}{}^{0}}\right)\left(\frac{q_{sg} / A_{sg}}{q_{sg}{}^{0} / A_{sg}{}^{0}}\right)^{-0.75} \cdot exp\left(\frac{P_s^{\,0} - P_s}{900}\right)$$

The fouling resistance ratio, $\frac{R_{foul}}{R_{tot}{}^{0}} = \frac{R_{foul}{}^{0}}{R_{tot}{}^{0}}$, is not a MARVEL-M input parameter, but is internally back-calculated within MARVEL-M as 1 – RRPF – RRTM – RRBO. Note also that the absolute values of these steam generator heat transfer coefficients at nominal conditions are calculated internally by MARVEL-M based on the user-input values of the ratios described above at nominal conditions.

RAI 2.1-9

Please provide the details of over-temperature ΔT and over-power ΔT trip protection methodologies and specifications, including uncertainties.

Response

MARVEL-M simulates both the overtemperature ΔT and overpower ΔT reactor trips. Although most MARVEL-M simulated reactor trip signals compare process variables with fixed setpoints to actuate the reactor protection system (RPS), the overtemperature ΔT and overpower ΔT reactor trip models use more complex setpoint algorithms. Section 2.3.2 of GEN0-LP-480 (the MARVEL-M code manual) provides a detailed description of the core operating limit protection (ΔT) trip models used in MARVEL-M. Revision 3 of GEN0-LP-480 was submitted to the NRC by MHI letter UAP-HF-08138 on August 1, 2008. A summary of the MARVEL-M code manual description is provided below.

Overtemperature ΔT

The overtemperature ΔT trip provides protection to prevent departure from nucleate boiling (DNB) and hot-leg boiling. In MARVEL-M the DNB and hot leg boiling limits are represented by conditions of the temperature difference across the vessel as a function of T_{avg} and pressure. When the reactor coolant loop ΔT exceeds the calculated $\Delta T_{setpoint}$, the reactor is tripped.

The following equation defines the trip setpoint for the DNB limit line:

$$\Delta T_{setpoint} = \Delta T^{nom}\left[K_1 + K_2\left(P - P^{nom}\right) - K_3\frac{1+\tau_3 s}{1+\tau_4 s}\left(T_{avg} - T_0\right) - f(\Delta q)\right]$$

where τ_3 and τ_4 are preset adjustable time constants to compensate for the preset manually adjustable bias; K_1, K_2, and K_3 are preset manually adjustable constants; T_o is the nominal value of T_{avg}; and f(Δq) is a penalty function of the flux difference between upper and lower nuclear instrumentation.

A similar overtemperature ΔT trip setpoint without the f(Δq) flux difference penalty term can also be defined for hot leg boiling limits using different numerical values of the adjustable constants and time constants. To allow for variations in plant RPS designs, the $\Delta T_{setpoint}$ for DNB and hot leg boiling limits can either be input as separate limits or expressed as the single equation shown above that bounds both limits. If the separate limits are modeled, the trip will occur when either setpoint limit is exceeded.

The overtemperature ΔT trip setpoint in MARVEL-M is selected such that under accidental conditions the trip would occur well within the DNB and hot leg boiling limits, even if all the adverse instrumentation setpoint errors are accumulated in the unfavorable direction. Time delays in signal measurement and processing are also included, which are approximated by a first order delay.

$$T_{avg} = \left[\frac{1}{(1+\tau_1 s)}\right]\cdot T_{avg-measured} \quad \text{and} \quad \Delta T = \left[\frac{1}{(1+\tau_2 s)}\right]\cdot\left[\frac{(1+\tau_6 s)}{(1+\tau_7 s)}\right]\Delta T_{measured}$$

where τ_1 and τ_2 are the delay time constant in the T_{avg} and ΔT measurement, respectively, and τ_6 and τ_7 are the lead/lag time constants in the ΔT measurement.

Setpoints for the DNB limit and the exit boiling limit are continuously and individually calculated by the RPS using a specific algorithm. Details of the US-APWR overtemperature ΔT setpoint algorithms are provided in DCD Section 7.2.1.4.3.1. DCD Figure 7.2-2 sheet 5 shows the US-APWR logic for this trip function. Although certain notation used in the DCD Chapter 7 description is slightly different, the functionality is the same as that modeled in MARVEL-M.

Overpower ΔT

The overpower ΔT trip provides protection against excessive core thermal power. In MARVEL-M the overpower ΔT trip setpoint is given by

$$\Delta T_{setpoint} = \Delta T^{nom}\left[K_4 - K_5\left(T_{avg} - T_o\right) - K_6\frac{\tau_5}{1+\tau_5 s}T_{avg} - f(\Delta q)\right]$$

where τ_5 is a preset adjustable time constant bias applied in the rate/lag operator for increasing average temperature; K_4, K_5, and K_6 are preset manually adjustable constants; and $f(\Delta q)$ is a function of flux difference between upper and lower nuclear instrumentation. The time delays in signal measurement and processing are the same as previously described for the overtemperature ΔT trip. Note that while the US-APWR reactor protection system has separate lead/lag units for both overtemperature and overpower ΔT input signals for T_{avg} and ΔT, the MARVEL-M code contains only a single array for lead/lag parameters. Therefore, it is the user's responsibility to assure that the lead/lag parameters associated with the appropriate trip signal are input to MARVEL-M for a given analysis.

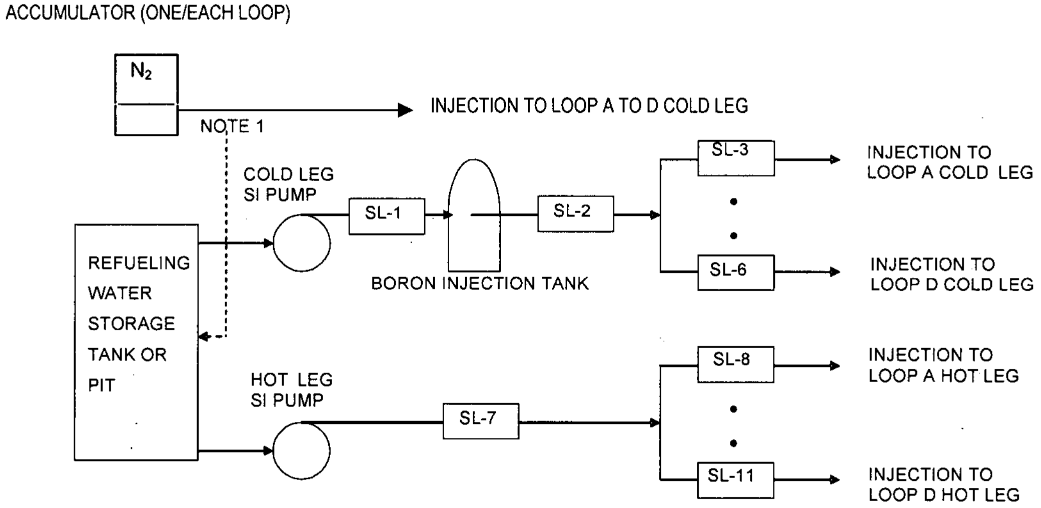
The setpoint for this trip is continuously calculated by the RPS using a specific algorithm. Details of the US-APWR overpower ΔT setpoint algorithm are provided in DCD Section 7.2.1.4.3.2. DCD Figure 7.2-2 sheet 5 shows the US-APWR logic for this trip function. Although certain notation used in the DCD Chapter 7 description is slightly different, the functionality is the same as that modeled in MARVEL-M.

RAI 2.1-10
Please provide the details of cold leg injection by the safety injection system and its role in non-LOCA events.

Response
The SIS modeling within MARVEL-M is briefly described on page 2-10 of MUAP-07010 with a more detailed description provided in Section 1.9.1 of GEN0-LP-480 (the MARVEL-M code manual). Revision 3 of GEN0-LP-480 was submitted to the NRC by MHI letter UAP-HF-08138 on August 1, 2008.

The US-APWR has four independent and dedicated safety injection (SI) pump trains which supply boric acid water from the refueling water storage pit (RWSP) directly to the reactor vessel downcomer in the normal system lineup.

As shown in the schematic drawing below from the MARVEL-M code manual, MARVEL-M has the capability to model several different safety injection system configurations. The upper flow path shown on the figure below models one SI pump, a common discharge header, and flow split to the four cold legs. This flow path model includes a Boron Injection Tank consistent with earlier plant designs. For newer plant designs such as the US-APWR that do not have a boron injection tank, this volume can be set to zero. Because for non-LOCA accidents the reactor coolant (RCS) pressure is the same at the cold leg inlets, one or more trains of SI pumps can be modeled with a single pump, as shown in the figure.


ACCUMULATOR (ONE/EACH LOOP)
N_2
INJECTION TO LOOP A TO D COLD LEG
NOTE 1
COLD LEG
SI PUMP
SL-1
SL-2
BORON INJECTION TANK
SL-3
INJECTION TO LOOP A COLD LEG
SL-6
INJECTION TO LOOP D COLD LEG
REFUELING WATER STORAGE TANK OR PIT
HOT LEG
SI PUMP
SL-7
SL-8
INJECTION TO LOOP A HOT LEG
SL-11
INJECTION TO LOOP D HOT LEG

Note 1: Blow down water from steam generators at secondary line break inside CV may enter the refueling water storage pit, if the storage pit is inside CV.

The lower flow path shown on the figure models a second single SI pump, a common discharge header, and flow split to the four hot legs. The user can select the injection point as the hot legs, the deluge line, or the reactor vessel downcomer.

A separate SI pump pressure-flow characteristic can be specified for each of the systems, and either of the systems can be turned on or off using user inputs. Initial boron concentrations and fluid enthalpies can be specified for the fluid in each of the SI piping volumes.

The SIS is credited in the safety analysis of only the following non-LOCA DCD events:

- inadvertent opening of main steam relief or safety valve,
- secondary steam system piping failure NSSS response,
- secondary steam system piping failure containment mass and energy release, and
- steam generator tube rupture.

The accumulator pressure is not reached for any of these non-LOCA events, so the accumulator model is not used.

For the first three steam flow increase events, the SI both provides RCS makeup to offset the RCS coolant shrinkage due to the cooldown and delivers borated water to the core for negative reactivity. In these events, it is important to accurately model the progression of boron to the core for the purpose of calculating core reactivity. For the steam generator tube rupture, SI is modeled for the purpose of calculating RCS inventory and pressure only. The boron reactivity coefficient is assumed to be zero for this event.

The cold leg injection (upper flow path) is used for all four of the above events for the US-APWR. Modeling considerations and boron transport models associated with the cold leg injection flow path for these events are described below.

For the steam release cooldown events, the time required for borated water to reach the core after SI initiation is determined by taking into consideration: (1) the period from the time the ECCS actuation signal is generated to the time the safety injection pumps reach full speed, and (2) the transport time for the injected water to pass through the SI piping and the reactor coolant piping. These delays and purge volumes are directly modeled in the MARVEL-M code. In the MARVEL-M model the high head ECCS injection cold leg injection point is conservatively assumed to be the cold leg node nearest the reactor vessel, although the actual injection point is the reactor vessel downcomer. The ECCS injection line is simulated in MARVEL-M considering the leading edge of the injection line water slug, which calculates the time delay associated with purging the low concentration borated water from the injection piping by the high concentration borated water from the RWSP. For the purpose of accurately tracking the progression of concentrated borated water through the SI piping, each of the piping volumes in the blocks labeled SL2 through SL6 are subdivided into 20 sub-volumes using slug flow models. The MARVEL-M model also includes the conservative assumption that no mixing occurs in the injection line nodes. Thus, the MARVEL-M SIS model is conservative from the point of view of the boron concentration of the core.

For the inadvertent opening of main steam relief or safety valve, several assumptions about the SIS are made. The boron concentration in the RWSP is assumed to be less than or equal to the minimum allowable Technical Specification value. The total pump flow is defined consistent with the assumptions in the DCD for operating pumps to inject the borated water from the RWSP. The time required for the borated water to reach the core is determined as discussed previously.

Similar assumptions about the SIS are made for the secondary steam system piping failures. Again, the boron concentration in the RWSP is assumed to be less than or equal to the minimum allowable Technical Specification value, the total pump flow is defined consistent with the assumptions in the DCD for operating pumps, and the time required for the borated water to reach the core is determined as discussed previously. Time delays for the start of the SI pumps are varied to account for whether the pumps are being powered by offsite power or standby emergency power.

For the steam generator tube rupture, the SI boration capability is not credited. Unlike the steam release cooldown events, SI flow is a penalty for the time to equalize the primary and secondary pressures. Maximum flow with all pumps delivering equally to all cold legs is assumed through user input of the system pressure-flow characteristic and uniform injection line split fractions. Because boron concentration is not of concern to the transient response, the boron reactivity coefficient is set to zero. The point of injection (cold leg vs. reactor vessel downcomer) does not affect the transient response for the steam generator tube rupture.

In summary, the cold leg injection flow path in the MARVEL-M code is used for all non-LOCA events, with event-specific inputs to conservatively model each event. The models for boron transport for the SI system volumes have special slug flow provisions that correctly follow the leading edge of borated water as it progresses to the core, resulting in the accurate prediction of core reactivity. These models and associated code inputs are further described in the MARVEL-M code manual. Assumptions made for each of the events crediting SI are described in their respective DCD analysis.

RAI 2.1-11

It is stated that the MARVEL-M algorithm for core mixing is changed from the original MARVEL model. Are the changes due to the expansion of the capability to four loops, or are there fundamental changes in the mixing phenomenology?

Response

The changes to the MARVEL-M algorithm for mixing in the reactor vessel are due to the expansion of the modeling capability from two to four loops. The basic assumptions are the same as the 2-loop version, as described in Section 2.1.3.2 of MUAP-07010, as repeated below.

1. Cross flow may only exist between each downcomer flow section and the adjacent downcomer sections.
2. Cross flow occurs so that coolant flow rates at the downcomer exit are uniform.
3. The cross flows are proportional to the differences of downcomer inlet flows.

From the above assumptions,

where

- Nloop = 4 for the US-APWR.
- W_i represents the portion of flow out of the downcomer node that flows directly into the corresponding reactor vessel lower plenum node for loop i.
- WCR_i represents the cross flow out of the downcomer nodes as defined in MUAP-07010 Figure 2.1-5 and determined by solving the above equations.

RAI 2.1-12

Mixing is assumed to occur in the reactor vessel lower plenum as specified in the code input. How are the mixing factors (FMXI) established by the user? What guidance is provided to the user?

Response

The mixing factor in the lower plenum (FMXI) is established using hydraulic test data and the equations described in the response to RAI 2.1-13. Standard 4-loop mixing factors were determined by MHI for the US-APWR and incorporated into a base MARVEL-M input file. Values for specific events are provided to the users in event-specific analysis procedures.

RAI 2.1-13

Some 1/7-scale mixing tests were carried out in the 1970's. It is suggested that mixing assumptions can be inferred from the published results. What justification can be provided to support these claims for evaluating FMXI? Please provide documentation of the 1/7-scale tests.

Response

MHI has utilized hydraulic test data from the 1/7-scale mixing tests for the[ ] reactor vessel model in order to determine the appropriate values of MARVEL-M input parameters FMXI and FMXO. Although the reactor internal design is different between the [ ] and the US-APWR (incore instrumentation moved to top of vessel, 14 foot core, etc.), the mixing factor itself has very little effect on the calculated DNBR as described in Appendix E of MUAP-07010 which shows a sensitivity analysis of DNBR as a function of the mixing factor for the steam line break event. An additional discussion of the Appendix E results is provided in the response to RAI App.E-1.

The MARVEL-M modeling of the reactor vessel mixing is described on pages 2-15 through 2-17 of MUAP-07010. A summary of the relevant variables and equations related to the calculation of the MARVEL-M input parameters FMXI and FMXO is provided below (excerpted from MUAP-07010).

The factor, f_m, is defined as the fraction of flow entering the reactor vessel from one loop that returns to the same loop upon exiting the core.

The factor f_{mi} is defined as the fraction of loop coolant flow which flows up the azimuthal sector (per loop) of the core nearest the inlet nozzle from which it emerges. The factor f_{mi} is related to the MARVEL-M input parameter FMXI, as shown below.

$$FMXI = (1 - f_{mi})\frac{Nloop}{(Nloop - 1)} \qquad (1)$$

where Nloop = 4 for the US-APWR

Similarly, the factor f_{mo} is defined as the fraction of the vessel outlet flow that comes from the azimuthal part (per loop) nearest the outlet nozzle. The factor f_{mo} is related to the MARVEL-M input parameter FMXO, as shown below.

$$FMXO = (1 - f_{mo})\frac{Nloop}{(Nloop - 1)} \qquad (2)$$

Additionally, f_{mi} and f_{mo} are related to f_m as follows:

$$f_m = f_{mi} \cdot f_{mo} + \frac{(1 - f_{mi}) \cdot (1 - f_{mo})}{(Nloop - 1)} \qquad (3)$$

MHI utilizes the following values of f_m and f_{mi} in order to calculate MARVEL-M input parameters FMXI and FMXO from the equations (1), (2), and (3) above. The design value of f_{mi} is utilized for all DCD Chapter 15 non-LOCA events for the US-APWR, except for the steam line break, which uses the conservative value of f_{mi}.

[ ]

The values for f_m and f_{mi} are taken from WCAP-7635 (Reference 1). These values are based on the report of the 1/7 scale mixing tests for the[ ]reactor vessel. The test result report is referred to in WCAP-7635, but is not publically available. Although the DNBR analyses are insensitive to the mixing factors, as described previously, MHI recently carried out a 1/7 scale mixing test for the US-APWR reactor vessel in order to confirm the applicability of the[ ]test results. The experimental results indicated that the mixing factors used in the safety analyses that were based on the[ ]test are still applicable for the US-APWR as described in MUAP-07022-P, "US-APWR Reactor Vessel Lower Plenum 1/7 Scale Model Flow Test Report" which was submitted on June 30, 2008 (ML081850466).

Reference (Attached to RAI response letter as Enclosure 4)

1) WCAP-7635, "MARVEL – A Digital Computer Code for Transient Analysis of a Multiloop PWR System, dated March 1971.

RAI 2.1-14

The methodology for reactor vessel upper plenum mixing is exactly the same as for mixing in the lower plenum. The mixing factor is FMXO. How are the mixing factors (FMXO) established by the user? What guidance is provided to the user?

Response

The mixing equations for FMXI (Eq. 13) and FMXO (Eq. 18) described in MUAP-07010 Section 2.1.3.2 use the same mathematical expression to describe the portion of the main flow that is mixed rather than directly transferred to the next flow node without undergoing mixing. However, the meaning of the mixing factors in the lower plenum and upper plenum is different.

FMXI represents the fraction of downcomer flow in the corresponding lower plenum node that comes from the other loops. Alternatively, 1-FMXO represents the fraction of flow leaving the reactor vessel upper plenum lower part that exits the reactor vessel by the nearest hot leg. Refer to Figure 2.1-5 in MUAP-07010.

The mixing factor in the upper plenum (FMXO) is established using hydraulic test data and the equations described in the response to RAI 2.1-13.

Standard 4-loop mixing factors were determined by MHI for the US-APWR and incorporated into a base MARVEL-M input file. Values for specific events are provided to the users in event-specific analysis procedures.

RAI 2.1-15

Are there any scaled experimental data to use for guidance in determining FMXO? If so, please provide the data or guidance.

Response

The response to RAI 2.1-13 describes how to calculate the mixing factors FMXI and FMXO based on the best available mixing test data.

RAI 2.1-17

Transition to natural circulation flow is modeled in MARVEL-M and the elevation head equations are typical. Has this model been tested against scaled experimental data? If so, please provide comparisons.

Response

The MARVEL-M transient flow model has not been compared to scaled experimental data. However, the MARVEL-M model has been compared with LOFTRAN (in a manner similar to the code comparisons provided in MUAP-07010 Section 3) in order to verify the natural circulation model.

Figure 2.1-17.1 shows the loop mass flow rate comparison between MARVEL-M and LOFTRAN for the "Loss of Non-Emergency AC Power to the Station Auxiliaries" event. The natural circulation flow of both the loop without emergency feedwater (EFW) and the loops with EFW after the reactor coolant pump (RCP) coastdown have very good agreement.

The US-APWR safety analysis assumes a conservative lower limit on the friction and windage torque utilized in the MARVEL-M RCP model. (See the response to RAI 2.1-7 for details on the RCP model.) The MARVEL-M results presented in Figure 2.1-17.1 do not include this conservative limitation in order to better compare the results with the LOFTRAN code. Figure 2.1-17.2 presents the results of the case where the safety analysis assumed friction and windage torque lower limit is used in MARVEL-M compared with the same LOFTRAN results previously shown in Figure 2.1-17.1. The figure demonstrates that the safety analysis MARVEL-M predicted results show a larger decrease in flow rate and a lower stable natural circulation flow level than predicted with LOFTRAN. In the DCD Chapter 15 analyses of the US-APWR, this conservative assumption is used in events that feature natural circulation flow conditions.

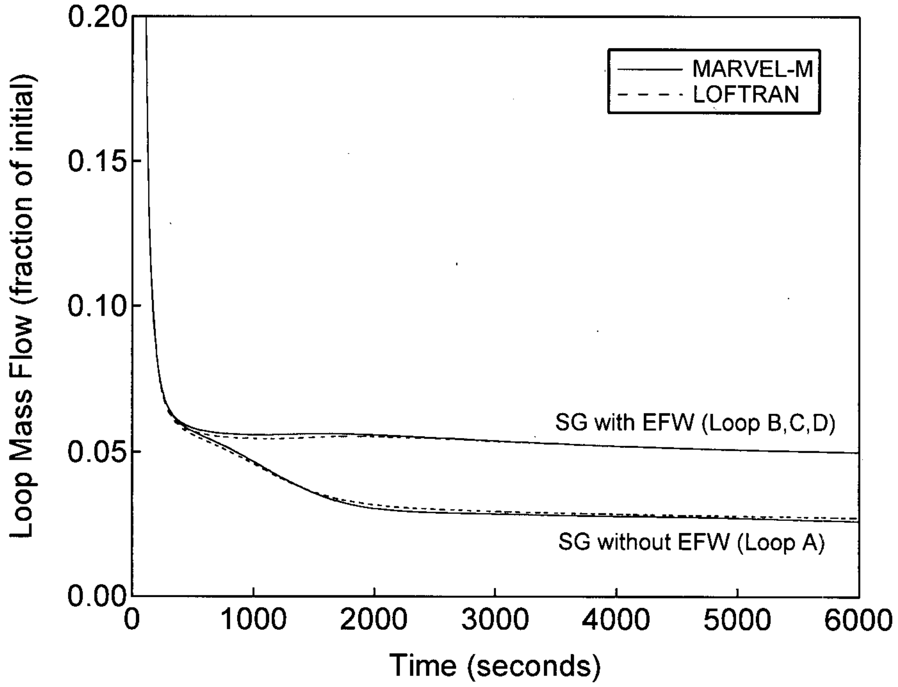


Figure 2.1-17.1 **Loss of Non-Emergency AC Power to the Station Auxiliaries Comparison between LOFTRAN and MARVEL-M (1)**

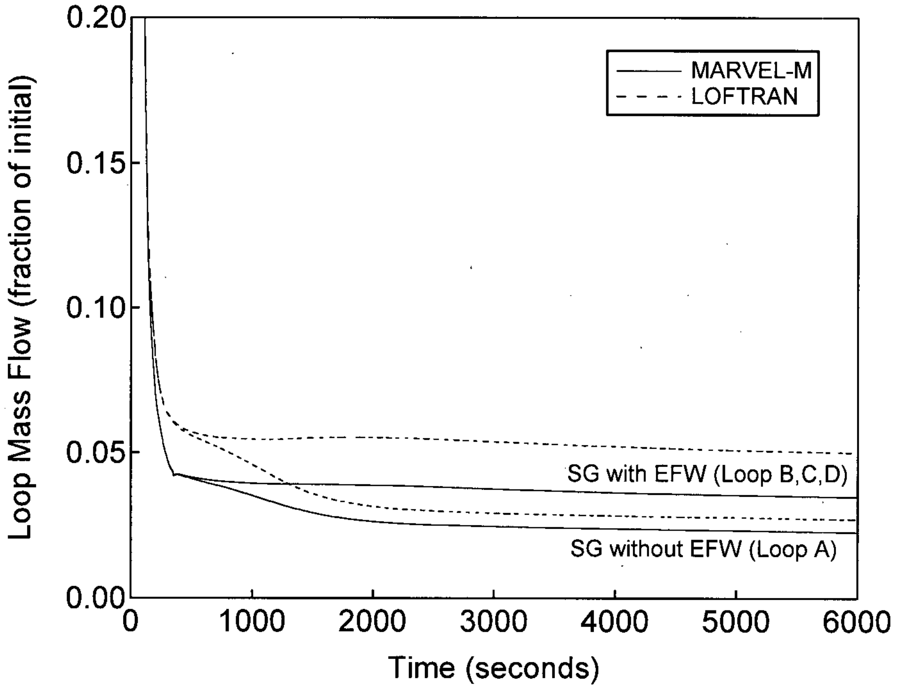


Figure 2.1-17.2 **Loss of Non-Emergency AC Power to the Station Auxiliaries Comparison between LOFTRAN and MARVEL-M (2)**

RAI 2.1-18

How does the transition to natural circulation depend upon the pump coast down model? Have sensitivity calculations been performed over a range of conditions to demonstrate confidence in the end state? If so, please provide the results.

Response

MARVEL-M now includes a reactor coolant pump (RCP) model so that flow transients can be explicitly computed by the pump model in conjunction with the existing reactor coolant system hydraulic models. Thus, there is no separate transition to natural circulation model because flow transient equations are solved at each time step depending on the status of the pump. Section 2.1.3.3 of MUAP-07010 provides a detailed description of the various equations utilized for modeling flow transients in MARVEL-M.

When the RCP in a reactor coolant loop is running, the head of the pump $[H_{PUMP}]_i$ is calculated from the flow coastdown equations [Equations (29) through (33) in MUAP-07010]. If some pumps are not running, the flow for the loop associated with the idle pump is reversed, and the pump head is replaced with a reverse flow pressure loss. When all the RCPs are not operating, all the pump heads, $[H_{PUMP}]_i$ are replaced with pressure losses and the reactor coolant flow transitions to natural circulation. The natural circulation flow in the multiple loops depends primarily on the power generation in the reactor core and the heat removal in the loops at the steam generators. The flow transition from the forced circulation to the natural circulation is calculated using Equation (36) in MUAP-07010. As a result, there is no need for an approximated model of the transition to natural circulation flow.

RAI 2.1-20

"Realistic models" have been added to MARVEL-M "to simulate real plant transient behavior." Although they are code options, they are not used for licensing evaluations of reactor plants. What controls are in place to ensure that this is the case?

Response

The realistic models available as code options in MARVEL-M are described in Section 2.1.4 of MUAP-07010 and Section 3.0 of GEN0-LP-480 (the MARVEL-M code manual). Details of the input data for MARVEL-M are provided in Part II of the MARVEL-M code manual. Revision 3 of this document was submitted to the NRC by MHI letter UAP-HF-08138 on August 1, 2008.

A standard base MARVEL-M input file has been created by MHI for the US-APWR design. This standard input file is modified for each accident according to the event-specific analysis procedures. Finally, the results of the event-specific calculation are documented in a calculation memo and verified by a qualified reviewer. These quality assurance controls assure that the realistic models are not improperly utilized for licensing evaluations.

RAI 2.2-1

There are several differences between TWINKLE and TWINKLE-M that are mentioned but not discussed in any detail. The introduction of more spatial points and discontinuity factors suggests that the numerical algorithm for solving the diffusion equations may have also been changed although it is not stated. Please list all changes and provide additional description of the differences between the codes.

Response

The solution methods and constitutive models of the TWINKLE-M code have not changed from the original TWINKLE code. MHI has modified several functions which are mainly concerned with the treatment of input data. A brief description of each change is provided below. GEN0-LP-517 Revision 0 (the TWINKLE-M Input Manual), submitted to the NRC by MHI letter UAP-HF-08138 on August 1, 2008, provides additional information regarding TWINKLE-M input.

(1) Spatial Mesh Expansion

In response to a change in the fuel failure thresholds for reactivity initiated events for Japanese LWRs in 1993, the maximum number of spatial mesh points in the TWINKLE-M code was expanded from 2000 meshes to a variable number in order to support a full core three-dimensional representation. The related variables in TWINKLE-M of fuel burnup, macroscopic and microscopic cross section, xenon distribution, fuel temperature, fast and thermal neutron flux, neutron velocities, and delayed neutron fractions were also expanded to accommodate the full core three-dimensional calculations. The capability to solve three-dimensional problems and solution algorithms were not changed.

(2) Introduction of a Discontinuity Factor

A discontinuity factor was added as a new input process in the program. The purpose of the discontinuity factor is to improve the representation of the local power distribution in the three-dimensional calculations. The addition of the discontinuity factor does not change the diffusion equations in TWINKLE. Instead, the discontinuity factor implementation is shown in Figure 2.2-1.1 and described as follows:

- The macroscopic cross section data is divided by the discontinuity factor before solving the diffusion equations (additional process).
- The neutron flux is solved using the unchanged TWINKLE diffusion equation subroutine.
- The neutron flux is divided by the discontinuity factor to determine the mesh averaged neutron flux (additional process).

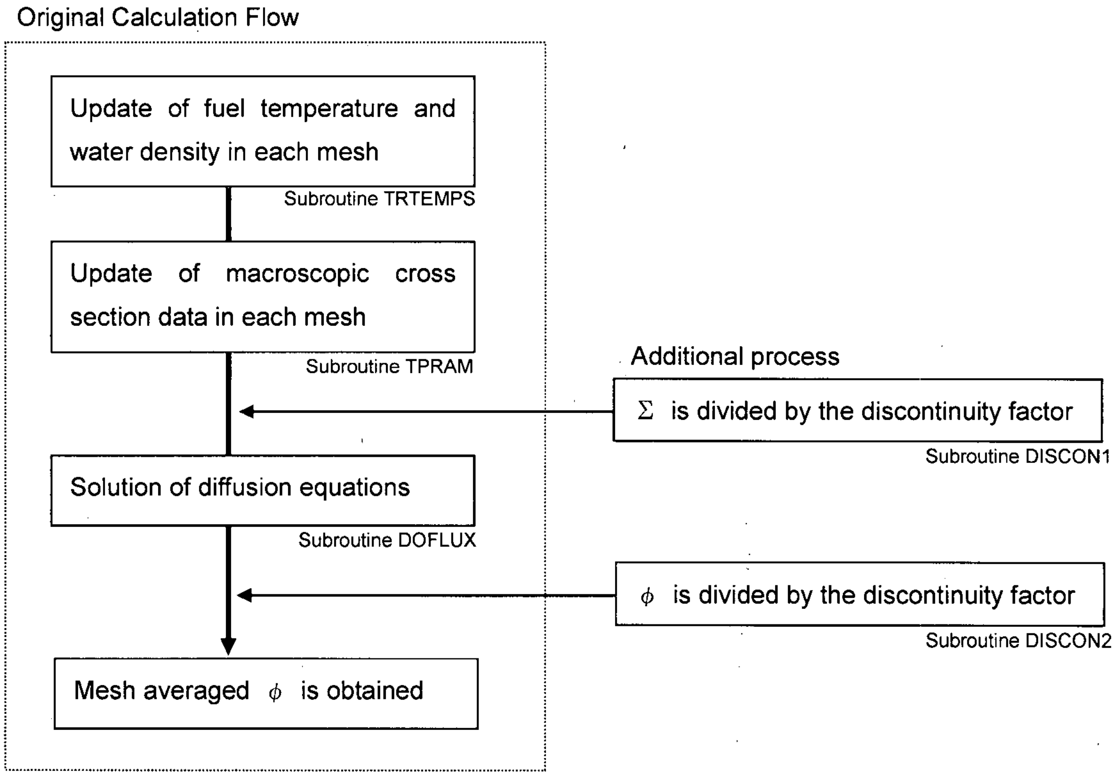

Original Calculation Flow
Update of fuel temperature and water density in each mesh
Subroutine TRTEMPS
Update of macroscopic cross section data in each mesh
Subroutine TPRAM
Additional process
Σ is divided by the discontinuity factor
Subroutine DISCON1
Solution of diffusion equations
Subroutine DOFLUX
ϕ is divided by the discontinuity factor
Subroutine DISCON2
Mesh averaged ϕ is obtained

Figure 2.2-1.1 Flowchart of the Discontinuity Factor Process

(3) Input Format
The input format was changed from a numerical identifier form to a Namelist form supported by standard Fortran 77 or 90 compilers. This allows users to identify the input list more clearly.

(4) Additional Options
The following features were added by MHI as input options:

- The nuclear and thermal properties of MOX fuel and BP were included.
- The number of fuel pellet divisions in the radial direction is expanded from 4 meshes to 10 meshes.
- Fuel properties depending on the fuel BU were added such as a thermal conductivity and a radial power depression in the pellet.
- Xenon distribution data calculated by ANC can be passed to TWINKLE-M through an input file. This change provides for a savings in calculation time.
- A separate equation was added to represent the dashpot portion of the trip curve. Prior to this change, the dashpot region was a continuation of the linear portion of the trip curve. The resulting trip curve is now an "S" shaped curve that is more representative of the RCCA displacement curve.
- New outputs were created to aid in the interpretation of code results by allowing for additional data plots and providing the ability to perform sequential calculations with VIPRE-01M. Specific examples of these changes include:
  - Time history of core average power, peaking factor, axial offset, etc.
  - Mesh-wise power distribution map for each time
  - Mesh-wise adiabatic fuel enthalpy rise during a power excursion

RAI 3.1-2

Please provide documentation of MARVEL-M/LOFTRAN code comparisons that may not be in agreement (if available), along with any explanations for the deviations.

Response

The overall agreement between MARVEL-M and LOFTRAN is generally quite good. This response describes a case where a difference between the two codes has been noticed, but can be explained based on known differences in the code models.

MUAP-07010 Section 3.1.1 provides a comparison between MARVEL-M and LOFTRAN for the Uncontrolled RCCA Bank Withdrawal at Power event assuming the pressure control system is turned off. While performing a similar comparison for the same case except with pressurizer pressure control, slight differences were observed in the pressurizer pressure response. After further review, the differences in RCS response for the LOFTRAN and MARVEL-M analyses were shown to be attributed to the following differences between the spray models in LOFTRAN and MARVEL-M:

MARVEL-M

- Saturated steam is assumed in the pressurizer
- If cold water is sprayed, the saturated steam quickly condenses

LOFTRAN

- Superheated steam is assumed in the pressurizer
- If cold water is sprayed, the spray first removes the super heat maintaining pressurizer pressure (the steam phase does not condense but shrinks)
- After sufficient water has been sprayed the pressurizer reaches a saturated condition and condensation of the saturated steam begins

Figures 3.1-2.1 through 3.1-2.4 show the comparison between MARVEL-M and LOFTRAN for the case where the pressure control system is available assuming the pressurizer spray is actuated by the actual pressurizer pressure signal. The LOFTRAN predicted RCS pressure is slightly higher than the MARVEL-M code prediction due to the previously described spray model differences. Because the differences occur below the safety valve pressure, differences in the safety valve models have been ruled out as the cause for the differences. In addition, as stated above, the transient comparison without pressurizer pressure control is virtually identical. The other transient parameters have good agreement between the two codes.

In the RCS pressure analysis, the pressure control system is conservatively turned off to avoid mitigating the pressure increase. Therefore, it can be concluded that MARVEL-M conservatively calculates peak pressures for events that challenge the reactor coolant pressure boundary acceptance criterion. For certain other events such as DNB events, it is conservative to assume the operation of the pressurizer pressure control system. In such cases, MARVEL-M with pressurizer spray operating is expected to calculate pressures slightly lower than LOFTRAN, resulting in a slightly conservative DNBR calculation. However, this effect is very small for the minimum DNBR compared with the other conservative assumptions and treatment of uncertainties for initial conditions for these DNB events. In addition, the minimum DNBR for the limiting Complete Loss of Flow event conservatively assumes constant RCS pressure in the VIPRE-M calculation.

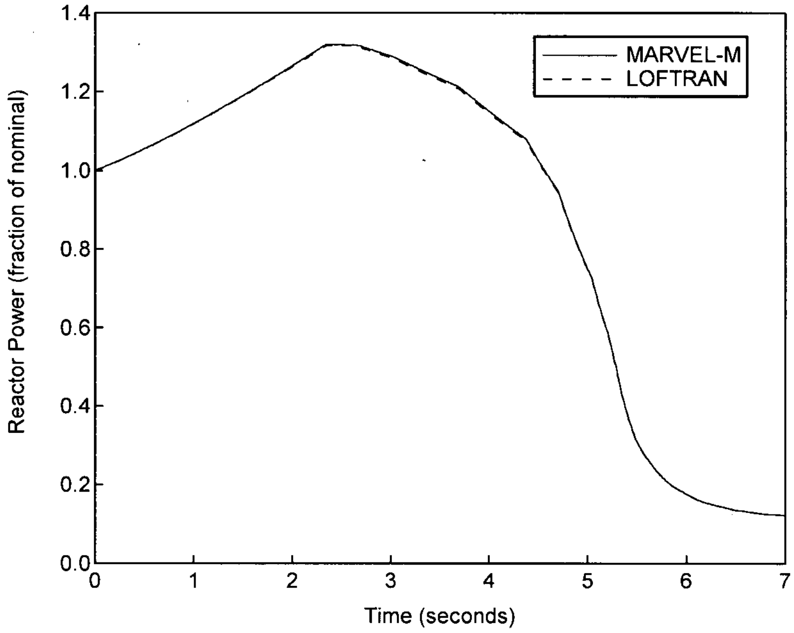


Figure 3.1-2.1 Reactor Power versus Time
Uncontrolled RCCA Bank Withdrawal at Power
Comparison between MARVEL-M and LOFTRAN

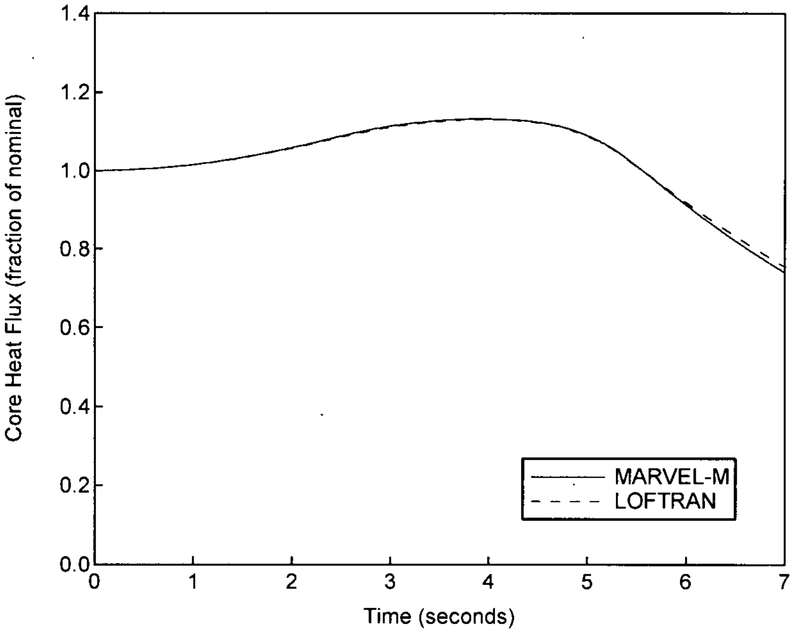


Figure 3.1-2.2 Core Heat Flux versus Time
Uncontrolled RCCA Bank Withdrawal at Power
Comparison between MARVEL-M and LOFTRAN

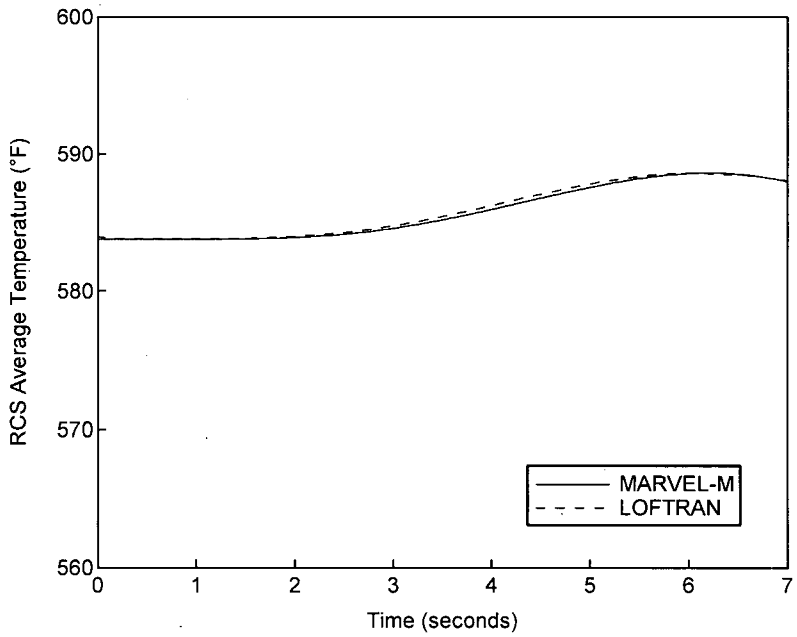


Figure 3.1-2.3 RCS Average Temperature versus Time
Uncontrolled RCCA Bank Withdrawal at Power
Comparison between MARVEL-M and LOFTRAN

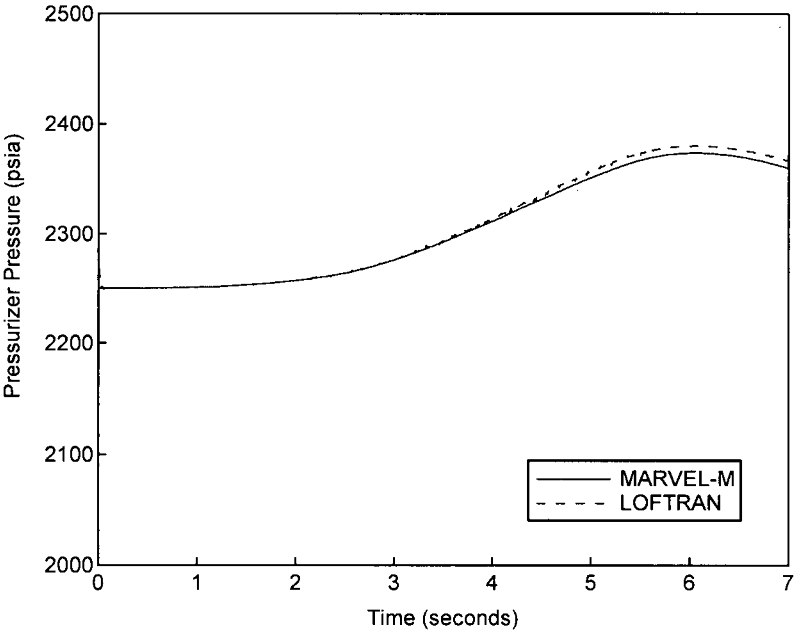


Figure 3.1-2.4 Pressurizer Pressure versus Time
Uncontrolled RCCA Bank Withdrawal at Power
Comparison between MARVEL-M and LOFTRAN

RAI 3.1-3

Please provide the DNBR vs. t for the partial loss of forced reactor coolant flow analyses for both MARVEL-M and LOFTRAN.

Response

As a clarification, DNBR is calculated by the VIPRE-01M code for this event using power and flow calculated by either MARVEL-M or LOFTRAN. The figure below provides a comparison of the DNBR for the partial loss of forced reactor coolant flow event. The two codes provide identical DNBR results for this event.

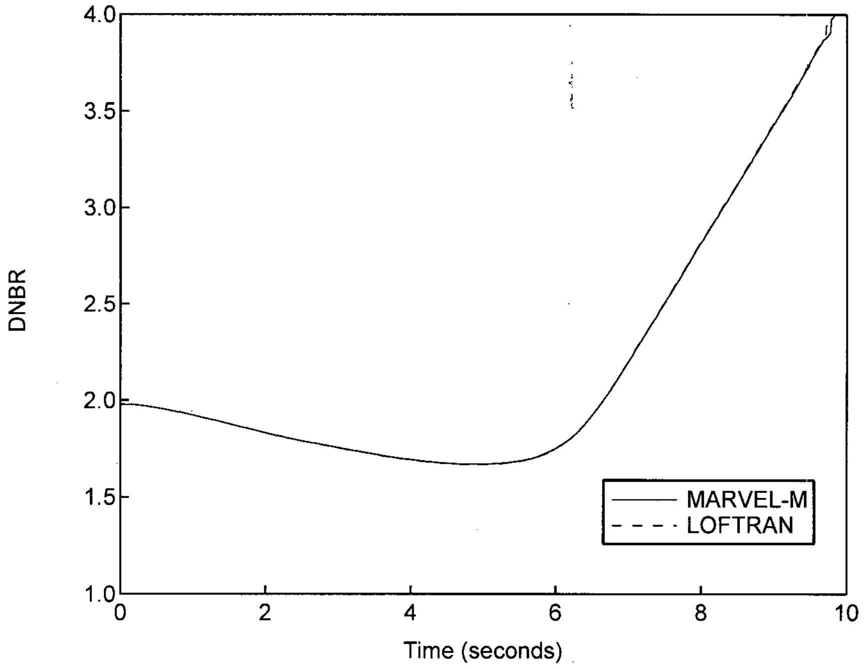


Figure 3.1-3.1 **DNBR versus Time**

Partial Loss of Forced Reactor Coolant Flow
Comparison between MARVEL-M and LOFTRAN

RAI 3.1-4

Please provide the DNBR vs. t for the complete loss of forced reactor coolant flow analyses for both MARVEL-M and LOFTRAN.

Response

As a clarification, DNBR is calculated by the VIPRE-01M code for this event using power and flow calculated by either MARVEL-M or LOFTRAN. The figure below provides a comparison of the DNBR for the complete loss of forced reactor coolant flow event. The two codes provide identical results for this event.

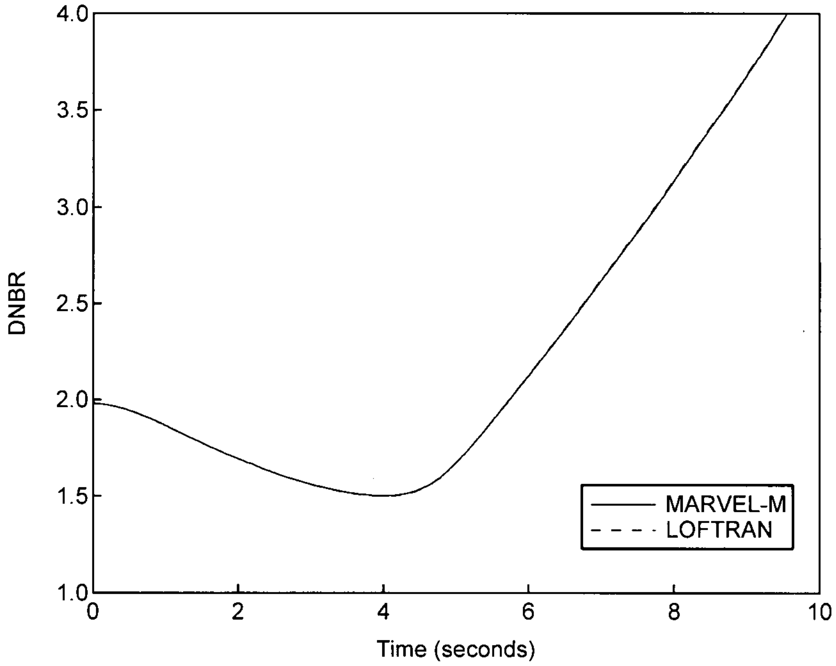


Figure 3.1-4.1 **DNBR versus Time**

Complete Loss of Forced Reactor Coolant Flow Comparison between MARVEL-M and LOFTRAN

RAI 3.1-5

Please provide the DNBR vs. t for the partial reactor coolant pump shaft seizure analyses for both MARVEL-M and LOFTRAN.

Response

As a clarification, DNBR is calculated by the VIPRE-01M code for this event using power and flow calculated by either MARVEL-M or LOFTRAN. The figure below provides a comparison of the DNBR results for reactor coolant pump shaft seizure event.

Note that the figure only shows the DNBR transient until the DNBR reaches its safety analysis limit. The two codes show identical results for this transient.

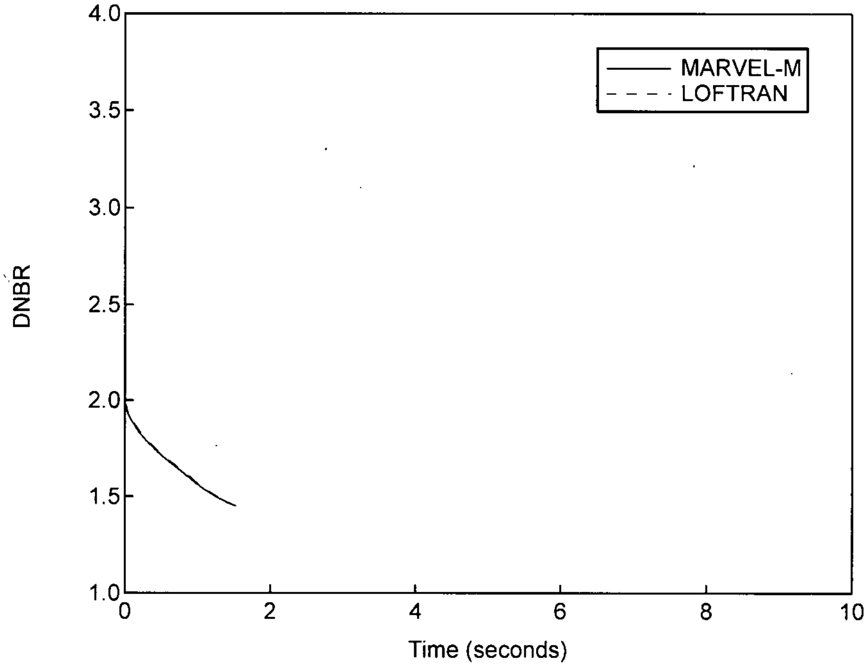


Figure 3.1-5.1 **DNBR versus Time**

Reactor Coolant Pump Shaft Seizure
Comparison between MARVEL-M and LOFTRAN

RAI 3.2-1

The text says that constitutive models have not changed in TWINKLE-M but the introduction of discontinuity factors can be considered as such. Please respond to this comment.

Response

The response to RAI 2.2-1 provides a detailed discussion of each significant model change between TWINKLE and TWINKLE-M, including the introduction of the new discontinuity factor input process in TWINKLE-M. As depicted in Figure 2.2-1.1 in RAI 2.2-1, MHI considers the addition of the discontinuity factor a pre-processing and post-processing activity outside the solution algorithm of the TWINKLE-M code. Therefore, the solution methods and constitutive models of the TWINKLE-M code have not changed.

RAI 3.2-3

The treatment of the core-reflector boundary condition is important especially when the ejected rod is near the core periphery. Please explain the algorithm by which the diffusion coefficient in the reflector is modified.

Response

The neutron flux distribution changes sharply in the core-reflector boundary which can be simulated accurately using a fine mesh finite difference method or a nodal method such as used in ANC. As an alternative to using such methods, the coarse mesh results can be adjusted to agree with static design code power distribution results in the core. This is done by adjusting the diffusion coefficient of the reflector region once with region-specific multipliers prior to running TWINKLE-M to assure that the TWINKLE-M power distribution agrees with the ANC results both before and after the rod ejection.

The process of modifying the diffusion coefficient is done externally to the code as part of defining the initial condition. Thus there is no change in the computational algorithm in the TWINKLE-M code.

RAI 3.2-4

The comparisons between TWINKLE-M and ANC at HZP show the largest differences in the assemblies with control rods inserted. What are the differences in modeling between the two codes that are claimed to be causing the larger errors?

Response

This difference can be attributed to differences in the numerical solution between TWINKLE-M and ANC. TWINKLE-M uses the finite difference method and ANC uses a nodal expansion method. As explained in the response to RAI 3.2-3, the nodal expansion method used in ANC more accurately calculates steep flux gradients such as near inserted control rods at HZP conditions. Therefore, the differences between TWINKLE-M and ANC in the assemblies with control rods inserted at HZP are larger than those of surrounding fuel assemblies. The differences between TWINKLE-M and ANC tend to decrease as the TWINKLE-M mesh size decreases. It is concluded that the differences between TWINKLE-M and ANC at HZP in assemblies with control rods are attributed to the different numerical solution methods.

RAI 3.2-5

The largest differences between TWINKLE-M and ANC for the HFP case cannot be due to the effect of modeling control rods as all rods are withdrawn. Please comment on the cause of the differences for the HFP case.

Response

This difference can be attributed to differences in the numerical solution between TWINKLE-M and ANC. TWINKLE-M uses the finite difference method and ANC uses a nodal expansion method. As explained in the responses to RAI 3.2-3 and RAI 3.2-4, the nodal expansion method used in ANC more accurately calculates steep flux gradients such as between fresh fuel assemblies and adjacent fuel assemblies in the peripheral region of the core. Therefore, the differences between TWINKLE-M and ANC for adjacent assemblies with large differences in burnup are larger than other combinations of adjacent fuel assemblies. The differences between TWINKLE-M and ANC tend to decrease as the TWINKLE-M mesh size decreases. Therefore, similar to the responses to RAI 3.2-3 and RAI 3.2-4, the reason for the difference between the TWINKLE-M and ANC results is the difference in numerical solution methods.

RAI 3.2-6

Table 3.2.1-1 provides a comparison between TWINKLE-M and ANC for key parameters. A key parameter for the REA that can be calculated with the two codes is the Doppler reactivity coefficient. Please provide a comparison of the Doppler coefficient for at least one core configuration.

Response

The Doppler reactivity coefficient is a key parameter in the control rod ejection event. The response to RAI 5.3-8 describes how the Doppler effect is adjusted in the safety analysis evaluations. A comparison of the Doppler temperature coefficient calculated using both TWINKLE-M and ANC is provided for an end of cycle (EOC) core at hot zero power just prior to the control rod ejection. The comparison between TWINKLE-M and ANC shows good agreement, as the calculated Doppler temperature coefficient is identical for both codes.

Table 3.2-6.1 Doppler Temperature Coefficient Comparison with ANC and TWINKLE-M

Code	Doppler Temperature Coefficient
ANC	1.9 pcm/°F
TWINKLE-M	1.9 pcm/°F

RAI 3.2-8

It is not clear why the neutron lifetime is given in Table 3.2.2-1. If it is meant to be a given reactor condition then how does it enter into the calculation? Whether it is a reactor condition or the result of an edit from the calculation, how is it obtained?

Response

Table 3.2.2-1 provides the core conditions (initial power, average coolant temperature, RCS pressure, ejected worth, delayed neutron fraction, and neutron lifetime) that must be satisfied in order to create a set of TWINKLE-M base cases that represent identical core conditions so that the effects of only changes in the mesh size can be evaluated in Section 3.2.2 of MUAP-07010. Neutron lifetime is an output parameter calculated based on volume averaged power weighted neutron velocity. This is a useful core parameter in characterizing the core response to power excursion events. Per the nuclear design group, the output value of this parameter for both the 2 x 2 mesh case and the 4 x 4 mesh case should be 8.0 microseconds. Adjustments were made to the base model to obtain this value as indicated for both cases in Table 3.2.2-1. Thus, once appropriately adjusted, this parameter does not affect the results of the sensitivity study of the mesh size for the RCCA ejection event.

RAI 3.2-9

How is the delayed neutron fraction calculated for use in TWINKLE-M? If a single number is used for all fuel assemblies, please explain how results would compare if using a number generated for each assembly.

Response

Table 3.2.2-1 provides the core conditions (initial power, average coolant temperature, RCS pressure, ejected worth, delayed neutron fraction, and neutron lifetime) that must be satisfied in order to create a set of TWINKLE-M base cases that represent identical core conditions so that the effects of only changes in the mesh size can be evaluated in Section 3.2.2 of MUAP-07010. The fraction of the six delayed neutron groups is calculated in TWINKLE-M separately for each mesh based on the local burnup. The delayed neutron fraction shown in Table 3.2.2-1 is provided as a core average value calculated by power weighting. This is a useful core parameter in characterizing the core response to power excursion events. TWINKLE-M uses the value for each mesh rather than the core average value in its calculations. Per the nuclear design group, the output value of this parameter for both the 2 x 2 mesh case and the 4 x 4 mesh case should be 0.44%. Adjustments were made to the base model to obtain this value as indicated for both cases in Table 3.2.2-1. Thus, once appropriately adjusted this parameter does not affect the results of the sensitivity study of the mesh size for the RCCA ejection event.

RAI 5.2-1

Are the calculated results for the complete loss of reactor coolant flow AOO sensitive to the reactor coolant pump coast down assumptions? Have any sensitivity calculations been performed?

Response

As described on page 5-5 of MUAP-07010, a conservative reactor coolant pump (RCP) flywheel moment of inertia is assumed for the analysis of the complete loss of forced reactor coolant flow. A sensitivity calculation was performed using the design moment of inertia of the RCP flywheel rather than the conservative value assumed in the topical report analysis. The results of the sensitivity calculation are shown in Figure 5.2-1.1 to Figure 5.2-1.6. The reactor power and hot channel heat flux are identical as can be seen in Figure 5.2-1.1 and Figure 5.2-1.2. The conservative RCP coast down assumption results in conservative values of the reactor coolant system (RCS) total flow, RCS average temperature, RCS pressure, and minimum DNBR, as can be seen in Figure 5.2-1.3, Figure 5.2-1.4, Figure 5.2-1.5, and Figure 5.2-1.6, respectively. It should be noted that the DNBR calculations for both cases assume the same fixed core inlet temperature for the duration of the transient.

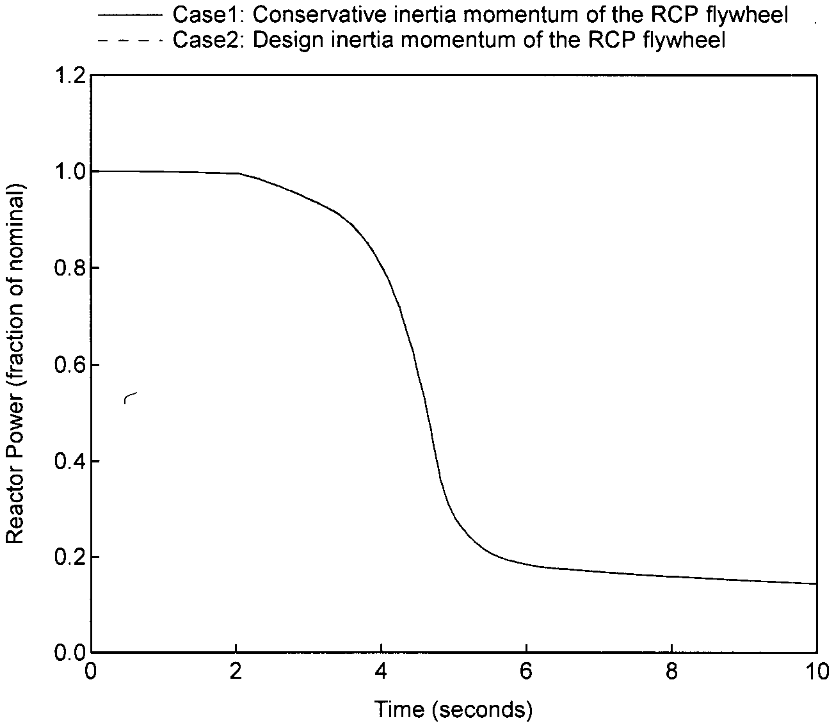


Figure 5.2-1.1 **Reactor Power versus Time**
Complete Loss of Forced Reactor Coolant Flow

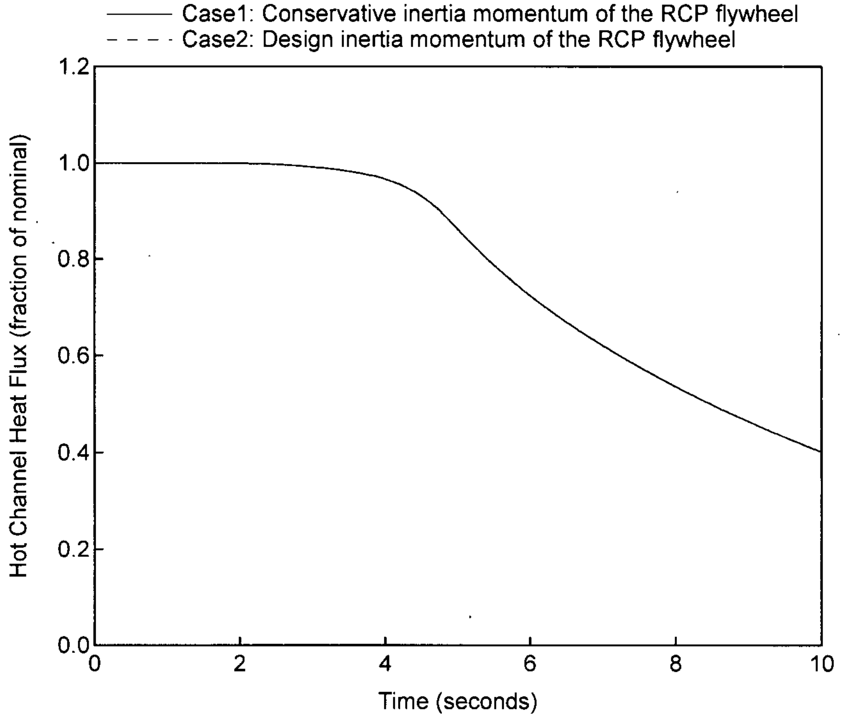


Figure 5.2-1.2 **Hot Channel Heat Flux versus Time**
Complete Loss of Forced Reactor Coolant Flow

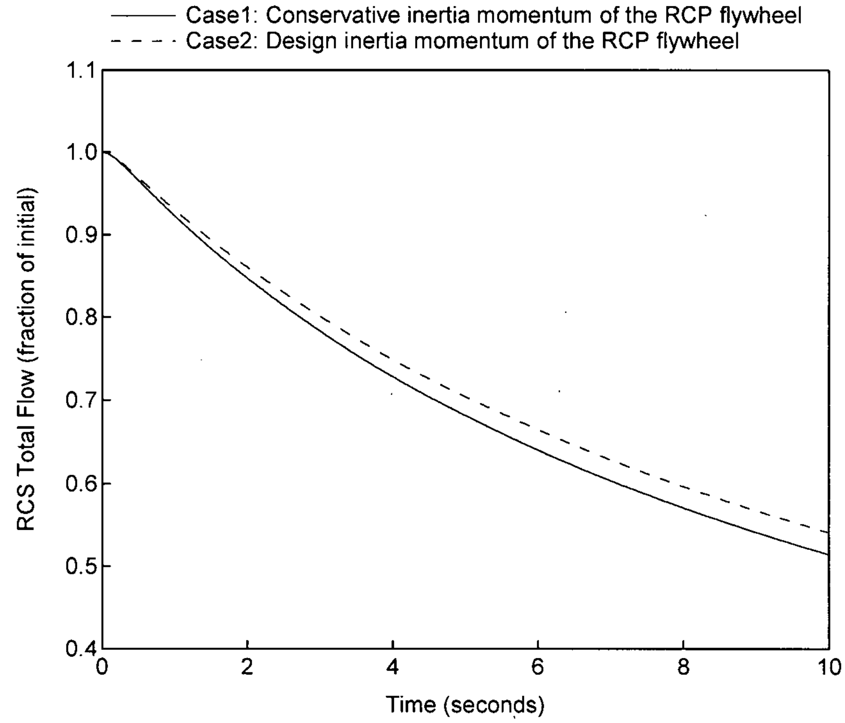


Figure 5.2-1.3 RCS Total Flow versus Time
Complete Loss of Forced Reactor Coolant Flow

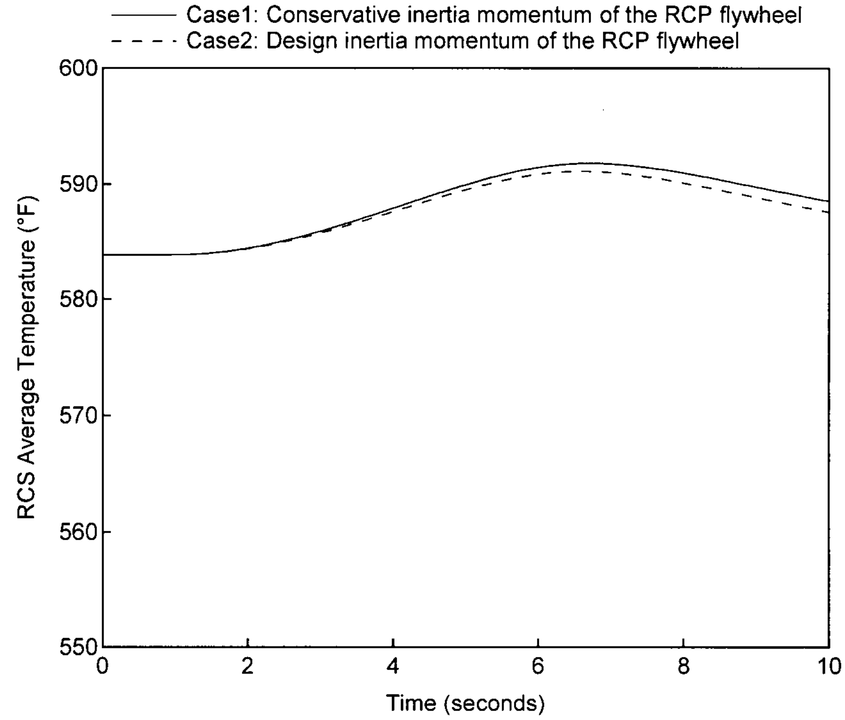


Figure 5.2-1.4 RCS Average Temperature versus Time
Complete Loss of Forced Reactor Coolant Flow

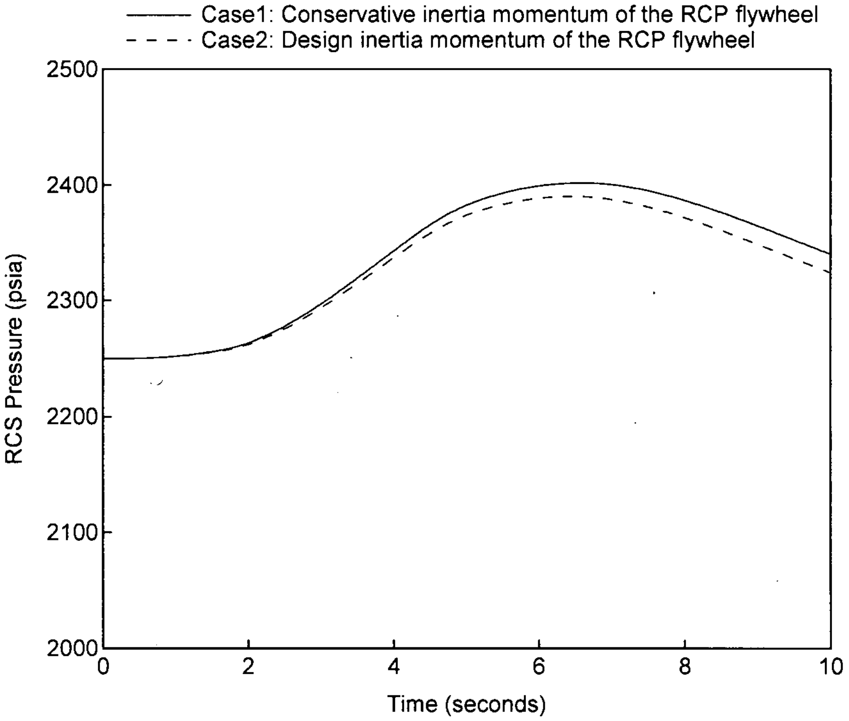


Figure 5.2-1.5 **RCS Pressure versus Time**
Complete Loss of Forced Reactor Coolant Flow

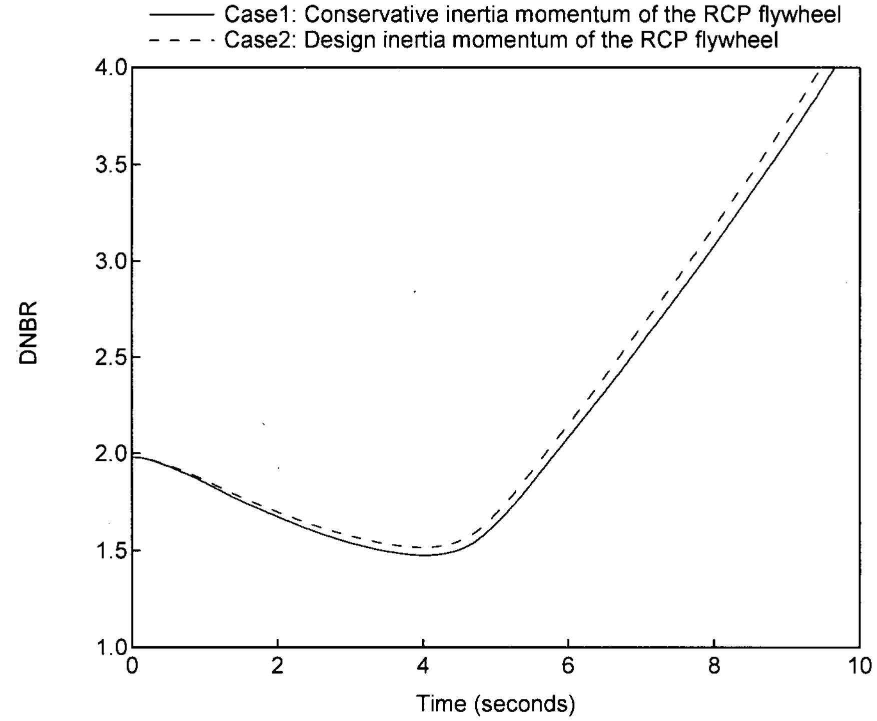


Figure 5.2-1.6 **DNBR versus Time**
Complete Loss of Forced Reactor Coolant Flow

RAI 5.2-2

For the complete loss of reactor coolant flow AOO, it is stated, "a suitable rod bundle DNB correlation...are used." Provide details of the DNB correlation.

Response

The US-APWR applies the WRB-2 correlation. The DNB correlation is intentionally not specified in this Topical Report on "non-LOCA methodology". The DNB correlations applicable to the Mitsubishi fuel design, which include WRB-2, are discussed in Topical Report MUAP-07009 "Thermal Design Methodology".

RAI 5.3-1

In the three- and one-dimensional analysis of the REA, a design limit for the ejected rod worth is used by adjusting the eigenvalue. How is the design limit determined? How is the eigenvalue changed in the calculation to simulate the ejection?

Response

As described on page 5-11 of MUAP-07010, a conservatively large reactivity, chosen at the design limit, is inserted within 0.1 seconds. The design limit for the ejected rod worth is determined as the maximum rod worth for different rod locations and core configurations using the static nuclear design code ANC.

In the case of the three-dimensional methodology which is used for the HZP case, the RCCA which has biggest ejected rod worth (most reactive RCCA) is selected. The inserted reactivity is directly simulated by the change in the absorption cross section caused by the ejection of the most reactive RCCA. The difference between the inserted reactivity and the design limit is accounted for by linearly changing the k_{eff} of the neutron kinetics equation over the course of 0.1 seconds.

In the one-dimensional methodology which is used for the HFP case, the design limit reactivity is externally added to the core only by linearly changing the k_{eff} of the neutron kinetics equation over the course of 0.1 seconds. The methodology in the HFP case differs slightly from the HZP case.

RAI 5.3-2

In the HZP analysis only the hot channel factor and core average power are passed on to VIPRE-M from TWINKLE-M. How is any detailed description of assembly powers and axial power distributions passed on to VIPRE-M? What is the advantage of using VIPRE-M when all the information needed to ascertain whether or not the acceptance criteria are met is contained in the TWINKLE-M results?

Response

The three-dimensional distribution of fuel enthalpy is calculated in TWINKLE-M using a mesh-wise average model, whereas the maximum fuel enthalpy rise is calculated in VIPRE-01M. The maximum enthalpy rise is calculated using a detailed sub-channel model in VIPRE-01M, which is, in turn, used to compensate for the difference between the mesh-wise and pin-wise enthalpy rise. The detailed procedure for how the three-dimensional distribution of enthalpy rise is adjusted to pin-wise enthalpy is provided in Section 5.3 of MUAP-07010.

To calculate the hot spot enthalpy rise in VIPRE-01M for PCMI failure, histories of core average power and hot channel peaking factor (F_Q) calculated in TWINKLE-M are passed to VIPRE-01M. The F_Q history is scaled using a multiplier so that the maximum value is the design limit, which is applied to the hot assembly of the 1/8 core model. The assumption of the power distribution around the hot assembly is the same as the thermal design power distribution and has no effect on the results; therefore, either a 1/8 core model or a single channel calculation can be used for this analysis. However, MHI has selected to use the 1/8 core model for the rod ejection analysis in order to assure consistency of the base input of VIPRE-01M with the core thermal hydraulic design.

In summary, to ascertain whether or not the PCMI acceptance criteria are met, the three-dimensional distribution of enthalpy calculated by TWINKLE-M is applied, considering a peak / average ratio in the mesh using the VIPRE-01M hot spot results. Histories of core average power and hot channel factor are necessary information to calculate the maximum enthalpy rise in VIPRE-01M.

RAI 5.3-3

Since hot channel design limits usually refer to a steady state condition, please explain the definition of the hot channel design limit for a transient.

Response

The hot channel design limit for a transient is determined using the static nuclear design code ANC and takes into account the calculation uncertainty and safety margin. Because the uncertainty and margin are not considered for the transient calculation by the TWINKLE-M code, the calculated maximum value of the hot channel factor history is lower than the design limit from ANC. This assumption is conservative for the core kinetics calculation because the lower hot channel factor decreases the local Doppler effect. On the other hand, for the fuel analysis using VIPRE-01M, the hot channel factor history is scaled by a multiplier so that the maximum value of the time-dependent hot channel factor history is equal to the design limit, which assures a conservative result for the hot spot thermal calculation.

RAI 5.3-4

Since only one hot channel factor (the maximum) is extracted from the TWINKLE-M calculation, how is it applied in VIPRE-M where an entire 1/8 core is represented? Why doesn't VIPRE-M do a single channel calculation?

Response

TWINKLE-M creates an interface file for use with VIPRE-01M that contains a time-dependent history of hot channel factor (not a single maximum value). In order to assure a conservative hot spot calculation, the hot channel factor versus time curve is adjusted upward so that the maximum value of the hot channel factor is equal to the design limit as calculated by ANC (see RAI 5.3-3 response and graphical representation provided in MUAP-07010 Figure 5.3-1). The assumptions of the power distribution in the hot assembly and the other assemblies are the same as the HZP case, as discussed in the response of RAI 5.3.-2. Note that the adjusted hot channel factor time history is only used for the three-dimensional calculation; the 1-D calculation assumes a constant hot channel factor.

The limiting parameters for the rod ejection event are fuel enthalpy, fuel temperature, and cladding temperature, which reach their respective maximum value before the transient coolant condition reaches and affects the core. Therefore, as the reviewer mentioned, a single channel calculation can be applicable for this analysis with an appropriate assumption of the hot spot power transient to be consistent with the peaking factor calculated by TWINKLE-M. However, MHI has selected to use the 1/8 core model for the rod ejection analysis in order to assure consistency of the base input of VIPRE-01M with the core thermal hydraulic design.

RAI 5.3-5

It is recognized that increasing the hot channel factor will make the calculation of fuel temperature and departure-from-nucleate-boiling ratio (DNBR) more conservative in the hot channel. However, assuming that the total power comes from the TWINKLE-M calculation, other channels will have lower powers in the VIPRE-M calculation relative to what was calculated in TWINKLE-M and this will make the calculation of temperature and DNBR in those channels lower. Since one acceptance criterion is a function of clad oxide thickness, the limiting channel may not be the hottest one. Is this taken into account by using the original TWINKLE-M distribution of fuel enthalpy rise rather than the distribution from VIPRE-M?

Response

As the reviewer mentioned, the original TWINKLE-M distribution of mesh-wise fuel enthalpy rise is used to ascertain whether or not the acceptance criteria are met. The hot spot enthalpy rise calculated in VIPRE-01M is used to compensate for the difference between mesh-wise and hot spot enthalpy rise. The detailed PCMI fuel failure evaluation methodology is provided in MUAP-07010 Section 5.3 (2) and Figure 5.3-1.

RAI 5.3-6

The VIPRE-M model uses a 1/8-core representation. This assumes symmetry that is not present in the neutronics calculation. However, this will be acceptable if there is no cross-flow out of that sector or if the time frame is too short for this to be important. Please comment.

Response

Section 4 of MHI topical report MUAP-07009 "Thermal Design Methodology" describes the VIPRE-01M core modeling scheme that coolant mixing between assemblies is conservatively ignored and the effect of mixing is limited to a very local area within the hot assembly. The power distribution in the hot assembly is assumed to be constant for the rod ejection analysis, which is conservative for the mixing effect. Additionally, the power of the hot assembly is assumed to be higher than the power of the surrounding assemblies, which would cause larger flow redistribution from the hot assembly to the surrounding assemblies and result in a more limiting coolant condition in the hot assembly. Therefore, the symmetric 1/8 core model is applicable for the rod ejection hot spot analysis.

RAI 5.3-7

Please explain how the TWINKLE-M one-dimensional model is obtained and how results for the axial power distribution compare with those obtained from a three-dimensional model.

Response

The evaluated state of the reactor core is analyzed with the static nuclear analysis code, ANC. The one-dimensional macroscopic cross sections for TWINKLE-M are created from the three-dimensional results of the ANC calculation. When the macroscopic cross section is collapsed from three dimensions to one dimension, the neutron flux and the volume of the node are used to weight each axial position in order to preserve the reaction rate in the radial direction. The weighting of the macroscopic cross section is shown in the following equation:

$$\overline{\Sigma_k} = \frac{\sum_{i,j} \Sigma_{i,j,k} \cdot \phi_{i,j,k} \cdot V_{i,j,k}}{\sum_{i,j} \phi_{i,j,k} \cdot V_{i,j,k}}$$

Where, $\overline{\Sigma_k}$ = Average macroscopic cross section of axial position k used with TWINKLE-M

$\Sigma_{i,j,k}$ = Macroscopic cross section of position i, j, and k used with ANC

$\phi_{i,j,k}$ = Neutron flux of position i, j, and k calculated with ANC

$V_{i,j,k}$ = Volume of node at position i, j, and k

The comparison of axial power distribution for the TWINKLE-M one-dimensional analysis and the three-dimensional analysis is shown in Figure 5.3-7.1. The analysis condition is beginning of the cycle, all control rods out, and hot full power (HFP). The analytical result of the TWINKLE-M one-dimensional analysis and the three-dimensional analysis agree very well as shown in Figure 5.3-7.1.

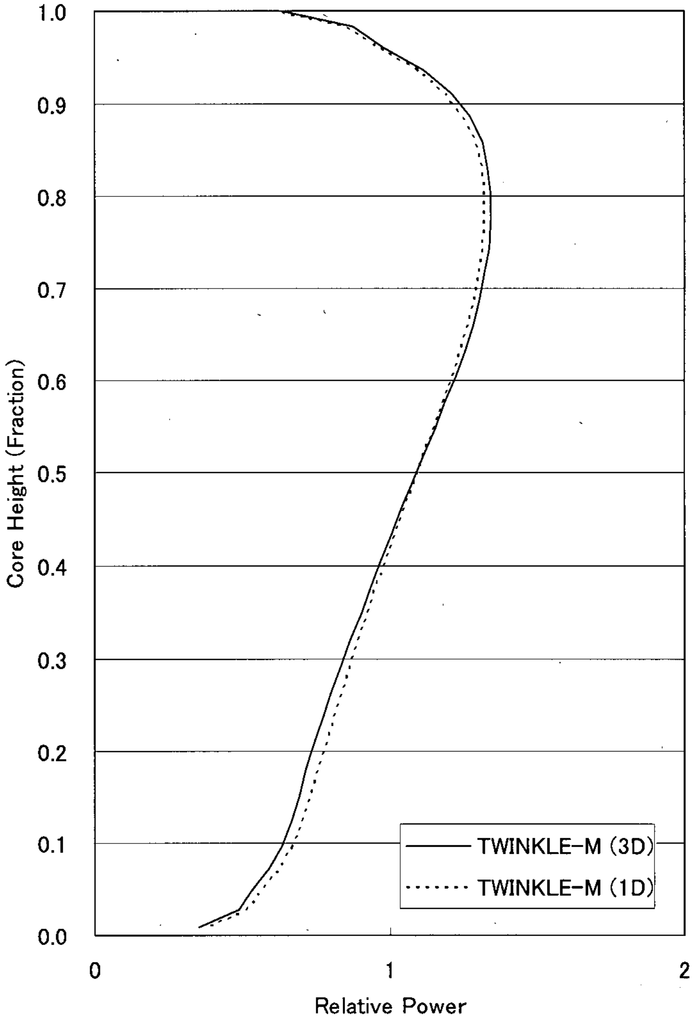


Figure 5.3-7.1 Comparison of Axial Power Distributions for BOC, HFP, ARO

RAI 5.3-8

What "conservative multiplier" is applied for the Doppler feedback?

Response

The Dopper feedback is applied as shown in the following equation. The parameter b is a constant used to adjust the fast absorption cross section for a given change in the calculated effective fuel temperature. In the safety analysis, the constant b is multiplied by 0.8 to apply 20% margin to the design value of the Doppler temperature coefficient.

$$\Sigma_{a1} = \Sigma_{a1}^{ref} + b \cdot \left(T_{eff}^{1/2} - T_{ref}^{1/2} \right)$$

where

Σ_{a1} = macroscopic absorption cross section for fast group

Σ_{a1}^{ref} = reference macroscopic absorption cross section for fast group

T_{eff} = effective fuel temperature

T_{ref} = reference fuel temperature

RAI 5.3-9

If a three-dimensional model is available in TWINKLE-M for doing the REA analysis at HZP, why isn't this same model used for the HFP case instead of shifting to a one-dimensional model?

Response

The main purpose of using the 3-D methodology is to properly capture the effect of Doppler feedback in the event where the power distribution is highly skewed, such as the RCCA ejection event. The power distribution is less skewed for the hot full power (HFP) case than it is for the hot zero power (HZP) case. As a result, there is not a large advantage in using the 3-D methodology for the HFP case. Thus, although it is technically possible to use the 3-D methodology for the HFP case, MHI opted to use the 1-D model in order to simplify the analysis.

RAI 5.3-10

For the modeling of reactor trip in TWINKLE-M it is stated that the reactivity insertion curve is "simulated." Does this mean that the reactivity insertion is not explicitly modeled by changing cross sections in fuel assemblies? If it is not explicitly modeled, how is the simulation done?

Response

The negative reactivity effect of the control rod is explicitly simulated by increasing the absorption cross section in the calculation mesh associated with inserting the control rod.

RAI 5.4-1

For the steam system piping failure, asymmetric power generation in the core occurs due to a non-uniform cool down. Is this accident sensitive to the user-input mixing factors for the lower and upper plena? Have any sensitivity calculations been performed? If so, please provide the results.

Response

The steam system piping failure event at hot zero power is a transient characterized by non-uniform cooling in combination with the assumption that the most reactive control rod is fully withdrawn. This event is sensitive to the mixing factors. Appendix E of MUAP-07010 evaluates the sensitivity of the calculated minimum DNBR to changes in the vessel mixing factors, FMXI and FMXO. Inlet mixing establishes the core reactivity for this event. The outlet mixing factor is a dependent parameter of f_m* and f_{mi}** as explained in RAI 2.1-13. The results of the analysis in Appendix E demonstrate that although less inlet mixing does result in small decreases in the calculated minimum DNBR, the DNBR remains significantly above the DNBR limit even assuming no inlet mixing.

* Fraction of flow entering the reactor vessel from one loop that returns to the same loop upon exiting the core.

** Fraction of loop coolant flow which flows up the azimuthal sector (per loop) of the core nearest the inlet nozzle from which it emerges.

RAI 5.4-2

It is stated "flow mixing in the reactor vessel is modeled in the code. The mixing factors for the reactor vessel inlet and outlet plena are defined conservatively by the input referring to the mixing test results by the 1/7 scale reactor vessel model." Please provide the model and substantiate why it is conservative.

Response

The response to RAI 2.1-13 provides details regarding the determination of the design value of f_{mi} used for all non-LOCA events except for the steam line break (SLB) and the conservative value of f_{mi} used for the SLB event. The conservative value of f_{mi} corresponds to the peripheral assemblies closest to the cold loop in the 1/7 scale test results. For the SLB event this maximum value is conservatively applied to the entire 1/4 core sector in MARVEL-M. The outlet mixing factor is a dependent parameter of f_m* and f_{mi}** as explained in the response to RAI 2.1-13.

* Fraction of flow entering the reactor vessel from one loop that returns to the same loop upon exiting the core.

** Fraction of loop coolant flow which flows up the azimuthal sector (per loop) of the core nearest the inlet nozzle from which it emerges.

RAI 5.5-1

For the feed water system pipe break, the natural circulation model in MARVEL-M is invoked. Are the calculated results sensitive to the timing of the transition to natural circulation flow? Also, are the reactor vessel inlet and outlet plena mixing factor the same during natural circulation as for the case of forced flow, and what values for the mixing factors are used?

Response

The responses to RAI 2.1-17 and RAI 2.1-18 discuss the treatment of natural circulation flow in MARVEL-M. There is no transition model for natural circulation flow; it is explicitly calculated, therefore, the results are not sensitive to the timing of the transition.

The same inlet and outlet mixing factors (FMXI and FMXO) are used for all flow conditions, the values of which are defined in the response to RAI 2.1-13. It is expected that during natural circulation conditions, the mixing at the lower plenum and the upper plenum will be close to perfect mixing. It is therefore conservative to use the mixing factors from the forced flow condition for the natural circulation condition. A sensitivity study of the feedwater system pipe break results to the assumed mixing factors will be provided in the response to RAI 5.5-2, which will be separately provided to the NRC as agreed upon during the June 26, 2008 conference call.

RAI 5.6-1

For the steam generator tube rupture, "the reactor coolant in the reactor vessel upper head dead volume may flash and form a steam phase at the top..." Does this condition have any effect on upper plenum mixing, and are there any feedbacks to the lower plenum and reactor inlet mixing conditions?

Response

Although it is possible for the upper head dead volume to flash and form a steam phase at the top of the vessel head this is not predicted to occur in the steam generator tube rupture analysis. The following information describes the MARVEL-M model that would be used to describe such as situation.

US-APWR has the design that the primary coolant in the reactor vessel upper head flows into the RV upper plenum lower part through a control rod cluster guide tube, and then it mixes with the flow from the reactor core. MARVEL-M simulates this flow path. If boiling occurs at the reactor vessel upper head, upper plenum mixing occurs as follows:

1. The volume expansion due to boiling displaces the primary coolant from the reactor vessel upper head (20-2) into the RV upper plenum lower part (21-1, 21-2, 21-3, 21-4). See Figure 2.1-5 in MUAP-07010.
2. The mass of the displaced primary coolant is distributed equally to each node of the RV upper plenum lower part.
3. The displaced primary coolant results in an increase of the primary coolant mass in the RV upper plenum lower part. The exit flow from the RV upper plenum lower part (*WPi)* also increases due to the flow contributions from the coolant from the core, core bypass coolant flow, and the additional coolant from the reactor vessel upper head.
4. *WPi* is distributed between the flow into the node of the RV upper plenum upper part (*WPi0*) and the flow to each loop hot leg (*WPii*). The distribution is based on the following equations:

( )

5. The primary coolant is mixed perfectly in the RV upper plenum upper part (22-1) and flows to each loop hot leg (*WUPi*).

Therefore, boiling at the reactor vessel upper head has no impact on the lower plenum and reactor inlet mixing conditions.

RAI 6.2-1

For the complete loss of reactor coolant flow AOO, please specify how uncertainties in the input parameters, trip set points, and calculated variables are included in the calculations.

Response

The revised thermal design procedure (RTDP) is used to statistically account for uncertainties in the input parameters such as reactor power, reactor coolant system (RCS) pressure, RCS flow rate, and RCS average temperature. Conservative assumptions for the trip simulation (trip setpoint, signal processing delays, trip reactivity curve, rod drop time) are used in the analysis. The conservative values for parameters such as the inertia of the flywheel, the pressure drop of the primary system, the moderator density coefficient, and the Doppler power coefficient are also used in the analysis. Additional details of the event-specific assumptions are provided in the DCD Chapter 15 analyses rather than in the Non-LOCA Methodology Topical Report.

RAI 6.3-1

In the discussion of the HFP rod ejection accident sample case, it is stated that ANC is used to obtain local peaking factors. No mention of the ANC code is found in the section on the rod ejection accident methodology. Please clarify.

Response

MUAP-07010 Chapter 5 "Event-Specific Methodology" focuses on MHI's general US-APWR non-LOCA analysis methodology which utilizes the following three principal computer codes.

MARVEL-M	plant transient analysis code
TWINKLE-M	multi-dimensional neutron kinetics code
VIPRE-01M	subchannel thermal hydraulics analysis and fuel transient code

The NRC approved nuclear design code ANC is used as a core simulator for the US-APWR design. ANC is a three-dimensional two group diffusion core calculation code based on nodal expansion method. ANC is used for the static evaluation of the peaking factor for the rod ejection accident and the radial power distribution for the steam piping failure event. ANC is one of several nuclear design tools that is related to the core design described in DCD Chapter 4 and related topical reports. These codes are not the central focus of the non-LOCA methodology and are therefore not included in topical report MUAP-07010.

RAI 6.4-1

Is the steam system piping failure event sensitive to the location of the break? Is a break between the steam generator and turbine the most limiting break location for all accident conditions?

Response

The US-APWR design includes both a main steam isolation valve and a check valve in each steam line immediately outside the containment upstream of the main steam header and lines to the turbine. (See MUAP-07010 Figure 5.4-1 for a graphical depiction.) Depending on the combination of break location and single failure assumed, the steam flow may be non-uniform (from only one steam generator) or uniform (all steam generators contribute to the break flow), and may either be automatically isolated or result in an uncontrolled blowdown from one steam generator.

Including single failures, the most limiting break location is inside the main steam line check valve allowing the uncontrolled blowdown of a single steam generator through the flow restrictor integral to the steam generator nozzle. A large double-ended break in the steam header outside the containment vessel will result in all steam generators blowing down until steam line isolation, which will terminate the event. Although the short-term release is greater, the immediate termination makes this case less limiting. As a convenience to the analyst in creating a bounding scenario, ignoring the check valve in the affected steam line results in both a large common blowdown prior to isolation and an uncontrolled single steam generator blowdown after isolation. This is the case analyzed in Section 6.4 of the topical report.

RAI 6.5-1

The analysis assumes that "all of the steam generators are at the steam generator water level low trip set point." Is the pressurizer water volume sensitive to perturbations in this assumption?

Response

The feedwater line break is assumed to occur concurrent with the low steam generator water level reactor trip signal resulting from the loss of feedwater flow assumed as a precondition. This means that all steam generator water levels are at the low steam generator water level setpoint when the break occurs. This conservative precondition minimizes the total steam generator inventory available to remove heat from the reactor coolant system and makes the reactor trip system response independent of the steam generator pressure and level dynamics of the feedwater line break prior to the reactor trip. This assumption is also conservative regarding the evaluation of the maximum pressurizer water volume.

RAI 6.6-1

For the steam generator tube rupture event, manual reactor trip is assumed at time = 900 seconds because operator action is required to recognize the event. Would the transient results be different if the timing of the manual reactor trip were varied, i.e., 1200 seconds and 1500 seconds, instead of 900 seconds? What was the basis for choosing 900 seconds?

Response

A steam generator tube rupture (SGTR) event initiates the PCMS-based N-16 alarm, which prompts the operator to manually trip the reactor using conventional emergency operating procedures. The N-16 alarm is expected to occur within the first 2 minutes of an SGTR regardless of the coolant activity. In addition, the N-16 alarm is a reliable alarm to immediately identify the event as a SGTR. The analysis assumes that the operators can detect the event within 900 seconds which represents significant time margin for a simple action in the main control room. The assumed 900 second operator action time period will be verified to be conservative during later stages of the plant licensing following development of and training on the procedures.

RAI App.A-1

Appendix A compares the calculated DNBR from MARVEL-M and VIPRE-01M (steady state) for two cases. The code results are identical in both cases. However, the MARVEL-M DNBR analysis uses look-up tables that were generated by calculations with VIPRE-01M. Although it is expected that the VIPRE-01 M model is the same as approved by NRC, this comparison does not validate the DNBR model in MARVEL-M for other transient and accident conditions. Please comment.

Response

As described in the response to RAI 2.1-2, the lookup table consists of DNBR, core inlet temperature, pressure, and core heat flux. The table is created by a VIPRE-01M steady state calculation under the condition of constant primary coolant flow rate and constant power distribution. Therefore, this table can be adopted for transient analysis in each AOO and PA for the condition where the primary coolant flow rate is constant and bounded by the applicable power distribution. MUAP-07010 Appendix A provides a representative comparison between the interpolation of the MARVEL-M DNBR lookup table and the original VIPRE-01M DNBR data for the Uncontrolled Control Rod Assembly Withdrawal at Power event. It is pointed out that the VIPRE-01M data points plotted in Figures A-1 and A-2 are not identical to the data entered into the MARVEL-M tables. This comparison simply demonstrates that the MARVEL-M interpolation methodology adequately calculates DNBR.

RAI App.E-1

Appendix E presents a set of sensitivity analyses based upon reactor inlet plenum mixing for the case of steam system piping failure. Although the case presented indicates that the assumption of no inlet mixing only causes a small reduction in the minimum DNBR, it cannot be extrapolated to other cases on this basis alone. Provide additional comparisons for other AOOs and PAs to substantiate the claim that DNBR is insensitive to the mixing assumptions.

Response

In assessing the importance and sensitivity of events to reactor vessel inlet mixing assumptions, it is important to understand that there is no sensitivity to reactor vessel inlet mixing for uniform events and for rapid transients initiated by the core or for rapid non-uniform transients that are terminated before the effects are felt by the core due to loop transit times. Based on this observation, events such as the RCCA ejection can be analyzed without a NSSS or reactor vessel model, and the partial and locked rotor & sheared shaft events are terminated before the effects of mixing reach the core. It should also be noted that uniformity in vessel inlet mixing is increased with decreasing reactor coolant pump flow and in the limit reactor vessel inlet mixing can best be approximated by perfect mixing for very low flows such as during natural circulation conditions. Perfect mixing is assumed for the Steam Generator Tube Rupture event due to natural circulation conditions that exist during most of the event.

As a result, the extent of reactor plenum inlet mixing has the largest effect on events characterized by non-uniform loop behavior, in particular, events resulting in the heatup or cooldown of a single reactor coolant loop. Among the events that result in a non-uniform loop heatup, the feedwater pipe break event is the most extreme. This event is assumed to be preceded by a uniform loss of feedwater as a precondition that reduces all steam generators to the low-low level trip setpoint, followed by the sudden loss of all inventory from one steam generator due to the break. This event is more extreme and more non-uniform than the loss of feedwater flow event where the non-uniform heatup is caused only by EFW failures assumed in the analysis. The response to RAI 5.5-2 will provide the sensitivity of the feedwater line break event to assumed mixing factors.

The steam system piping failure represents a limiting extreme for analyzing the effects of a single loop cooldown on the NSSS response. Again, by comparison, this non-uniform loop cooldown event is much more severe than the inadvertent opening of a steam generator relief or safety valve. This is the reason why the analysis provided in Appendix E for sensitivity of the event results to reactor vessel inlet mixing was presented for the largest steam system piping failure that results in an uncontrolled blowdown of a single steam generator.

As has already been discussed in the response to RAI 2.1-13, MHI assumes design mixing for all events in which the reactor coolant pumps are assumed to continue to operate, and in the extreme case of the main steam system piping failure, less mixing than expected by test results is assumed for this event. Notwithstanding the extreme non-uniform cooldown characterized by the large steam line break, Appendix E presents a sensitivity study for the steam system piping failure event and concludes that the DNBR would remain above the AOO 95/95 limit for this postulated accident with no mixing and in fact, is relatively insensitive to the mixing assumptions. Therefore, the conclusion that DNBR is relatively insensitive to the mixing assumptions for the most extreme non-uniform cooldown can be extrapolated to other cases.

RAI App.F-1

Please provide the Zaloudek correlation that MHI has used to perform the steam generator tube rupture break flow calculations.

Response

The Zaloudek correlation was originally developed for the critical flow of hot water through a short tube (Reference 1). The Zaloudek correlation is:

$$G = C\sqrt{\frac{2g}{v_f}\cdot\left(P_{up} - P_{sat}\right)} \qquad \text{for } (400 < P_{up} < 1800 \text{ psia})$$

where: G = mass flow rate per unit area for choked flow conditions
P_{up} = upstream pressure
P_{sat} = downstream (SG saturation) pressure
g = gravitational force constant
v_f = specific volume of saturated water
C = constant (0.95)

The Zaloudek correlation was then modified by MHI for use in defining the maximum initial steam generator tube rupture (SGTR) primary-to-secondary flow rate as follows. The constant "*C*" which was 0.95 in the original correlation was assumed to be 1.0. This assumption is conservative since the larger value of "*C*" will increase the mass flow rate. The density of saturated water was substituted for the specific volume of saturated water according to the relationship that $\rho_f=1/v_f$. For the SGTR, the saturation pressure corresponds to the saturation pressure of the secondary side. In the original correlation, the upstream pressure, P_{up}, corresponds very simply to the pressure of the RCS. However, in the SGTR analysis, the upstream pressure varies depending on the RCS pressure and location of the break along the steam generator tube. It must first be reduced by the pressure drop in the length of U-tube between the RCS and the break location. The pressure drop in a single U-tube under nominal RCS flow conditions can easily be determined. However, the pressure drop in the U-tube during the SGTR event will be much larger due to increased rate of flow due to the break, and is proportional to the square of the mass flow. The total pressure drop in the U-tube during the SGTR event can be expressed in terms of the nominal RCS flow and nominal pressure drop between the RCS and the break location by the expression:

$$\Delta P_{SGTR} = \Delta P_{nom}\cdot\left(\frac{G}{G_{nom}}\right)^2$$

where: ΔP_{nom} = pressure drop in the U-tube under nominal conditions
G = mass flow rate in the U-tube under the SGTR conditions
G_{nom} = mass flow rate in the U-tube under nominal conditions

From this expression it is clear that the pressure drop during an SGTR at high-pressure conditions is larger than the nominal pressure drop since the mass flow rate during the SGTR conditions is higher than the mass flow rate under nominal conditions. After the pressure drop in the U-tube has been determined, the upstream pressure can be defined as:

$$P_{up} = P - \Delta P_{SGTR} = P - \Delta P_{nom}\cdot\left(\frac{G}{G_{nom}}\right)^2$$

where: P = pressure in steam generator tube under nominal conditions (may be in hot or cold leg)

Substituting this expression into the original Zaloudek correlation, along with the assumptions described previously, yields the resultant correlation:

$$G^2 = 2g_c \cdot \rho_f \cdot \left\{ P - \Delta P_{nom} \cdot \left(\frac{G}{G_{nom}} \right)^2 - P_{sat} \right\}$$

where: G = SGTR break mass flow rate [lb/(ft^{2}s)]

G_{nom} = mass flow rate under nominal rated conditions [lb/(ft^{2}s)]

P = pressure in steam generator tube under nominal rated conditions (hot or cold leg) [lbf/ft^2]

ΔP_{nom} = pressure drop in U-tube to point of SGTR break under nominal rated conditions [lbf/ft^2]

P_{sat} = saturation pressure (secondary side) [lbf/ft^2]

g_c = gravitational force constant [(lb*ft/s^2)/lbf]

ρ_f= density of saturated water [lb/ft^3]

In this correlation the mass flow rate during the SGTR event, G, appears on both sides of the equation. Rearranging the terms, the "Modified Zaloudek" correlation used by MHI for calculating a conservatively high initial SGTR primary-to-secondary mass flow per unit area is the following:

$$G = \sqrt{\frac{2g_c \cdot \rho_f \cdot (P - P_{sat}) \cdot G_{nom}^2}{2g_c \cdot \rho_f \cdot \Delta P_{nom} + G_{nom}^2}}$$

Reference

1) Zaloudek, R. R., 1963, "The critical flow of hot water through short tubes", HW-77594, Richland, WA: Hanford Lab.